NEI 12-16, Revision 1

Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants

April 2014

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Nuclear Energy Institute

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FOREWORD

This guidance describes acceptable methods that may be used by industry to perform criticality analyses for the storage of new and spent fuel at light-water reactor power plants, in compliance with 10 CFR Part 50. The guidance provided herein is applicable to new fuel assemblies stored in a new fuel vault, and to new and spent fuel assemblies stored in a spent fuel pool.

Criticality requirements for the spent fuel pool of nuclear power plants are found in 10 CFR Part 50, Paragraph 10 CFR 50.68, or 10 CFR 70.24. Guidance for performing criticality analyses in compliance with these regulations were originally developed in a 1998 Nuclear Regulatory Commission internal memorandum by L. Kopp, and supplemented by the Standard Review Plan, NUREG-0800, Sections 9.1.1 and 9.1.2. Additional guidance was issued in an Interim Staff Guidance (DSS-ISG-2010-01) in 2011. This industry document is developed as a comprehensive guide that presents an acceptable approach to comply with the regulations, and upon NRC endorsement would supersede previous guidance documents. Individual vendors or licensees can deviate from the method supplied herein, with appropriate justification and approval by the NRC.

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ABBREVIATIONS AND ACRONYMS

AEG	Average Energy Group Causing Fission
APSR	Axial Power Shaping Rod
B&W	Babcock & Wilcox
BMU	Burnup Measurement Uncertainty
BPRA	Burnable Poison Rod Assembly
BWR	Boiling Water Reactor
CE	Combustion Engineering
CFR	Code of Federal Regulations
EALF	Energy of the Average Lethargy Causing Fission
ENDF	Evaluated Nuclear Data File
EPRI	Electric Power Research Institute
FTF	Fuel Transfer Form
GWD	Giga-Watt Days
IFBA	Integral Fuel Burnable Absorber
ISG	Interim Staff Guidance
LAR	License Amendment Request
MOX	Mixed-Oxide
MTU	Metric Ton Uranium
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Co-Operation and Development
ORNL	Oak Ridge National Laboratory
PWR	Pressurized Water Reactor
QA	Quality Assurance
RSS	Root Sum Square
SCCG	Standard Cold-Core Geometry
SFP	Spent Fuel Pool
SNM	Special Nuclear Material
WABA	Wet Annular Burnable Absorber

1 INTRODUCTION

1.1 PURPOSE

This document provides acceptable methods for performing criticality analyses for light-water nuclear reactor spent fuel pool storage racks and new fuel vaults. This guidance is applicable to both Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) facilities. These analyses are integral to the technical foundation for the design of nuclear fuel storage structures, systems and components, and the associated Technical Specifications in applications (i.e., License Amendment Requests (LARs)) submitted to the U.S. Nuclear Regulatory Commission (NRC) for review and approval.

This document is developed to provide comprehensive and durable guidance to improve consistency and clarity for performing criticality analyses that assure criticality safety and regulatory compliance. It is envisioned that this guidance will be endorsed by the NRC through a Regulatory Guide, which will achieve durability through NRC concurrence, and at such time this document will supersede previous guidance on criticality analyses for LWR facilities.

1.2 BACKGROUND

10 CFR 50.68 [1] was promulgated in 1998 to provide an analysis based alternative to the criticality monitoring required by 10 CFR 70.24 [2]. Prior to the rulemaking, exemptions to the monitoring requirement in 10 CFR 70.24 [2] was granted on a case-by-case basis for licensees demonstrating subcriticality through analysis. Compliance with either regulation is consistent with 10 CFR 50, Appendix A, General Design Criteria 62, "Prevention of Criticality in Fuel Storage and Handling." [3] 10 CFR Part 52 [4] was originally promulgated in 2007, and requires compliance with 10 CFR 50.68 [1].

The first guidance on acceptable methods for performing criticality analyses at LWR plants, following promulgation of 10 CFR 50.68 [1], was issued in 1998 through an NRC internal memorandum from L. Kopp to T. Collins, often referred to as the "Kopp Memorandum" [24]. Although this was an internal NRC memorandum, because of the lack of alternative formal guidance, it was quickly adopted by industry for use in performing criticality analyses, referenced in LARs, and referred to by NRC staff in the Safety Evaluation Reports for the associated license amendments. The guidance in the Kopp Memorandum provided regulatory clarity and stability for many years. In 2010, the NRC issued an Action Plan to develop new interim staff review guidance followed by a durable Regulatory Guide that would replace the Kopp Memorandum, and would better reflect the staff positions on acceptable criticality analysis methods that evolved through interactions with licensees since 2005.

NRC Interim Staff Guidance (ISG) DSS-ISG-2010-01, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," [25] was issued in 2011 to provide additional guidance to staff for the review of spent fuel pool storage rack criticality analyses. The guidance in DSS-ISG-2010-01 [25] is useful to support NRC staff review of industry criticality analyses until the more permanent and durable guidance in NEI 12-16 is endorsed by the NRC, at which time NEI 12-16 would supersede all previous guidance documents.

1.3 APPLICABLE REGULATIONS

The following regulations are applicable to criticality analyses for nuclear fuel storage at LWR facilities:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix A, General Design Criteria for Nuclear Power Plants Criterion 61, “Fuel Storage and Handling and Radioactivity Control.” [5]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix A, General Design Criteria for Nuclear Power Plants Criterion 62, “Prevention of Criticality in Fuel Storage and Handling.” [3]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.” [6]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.68, “Criticality Accident Requirements.” [1]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, “Technical Specifications.” [7]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47(a)(17), “Contents of applications; technical information.”; 52.79(a)(43), “Contents of applications; technical information in final safety analysis report.”; 52.137(a)(17), “Contents of applications; technical information.”; and 52.157(a)(8), “Contents of applications; technical information in final safety analysis report.” [4]

It is noted that in addition to the applicable regulations, the NRC has developed associated staff review guidance associated with the criticality analyses for nuclear fuel storage at LWR facilities.

- NUREG-0800, Standard Review Plan, Section 9.1.1, “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” Revision 4. [11]

- NUREG-0800, Standard Review Plan, Section 9.1.2, “New and Spent Fuel Storage,” Revision 3. [12]

1.4 DOUBLE CONTINGENCY PRINCIPLE

The double contingency principle, as described in ANSI/ANS 8.1, Section 4.2.2 [9], states that “process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible”. In other words, the nuclear criticality analysis is required to demonstrate that criticality cannot occur without at least two unlikely, independent and concurrent incidents or abnormal occurrences. This will ensure that no single occurrence can lead to a criticality. The double contingency principle means that a realistic condition may be assumed for the criticality analysis in calculating the effects of incidents or abnormal occurrences. When applying the double contingency principle, the conditions chosen need to be independent from one another (i.e. do not result from a common initiator) and are unlikely (i.e. low probability). For example, for PWRs, the loss of soluble boron below the Technical Specification limit is considered as one accident condition and a second concurrent accident need not be assumed (i.e. such as a fuel assembly misloading or misplacement). Therefore, credit for the presence of the Technical Specification soluble boron may be assumed in evaluating other accident conditions.

1.5 USE OF PRECEDENT

The use of precedent (i.e., adopting methods or conclusions previously approved in another application, but not documented in a generic regulatory document) is a well-established principle by the NRC in the process of reviewing applications. The use of precedent provides regulatory stability and efficiency. In order for a licensee to use precedent in an application, the licensee should demonstrate the applicability to its site specific analysis reflecting an evaluation of the similarities and differences from the original use. Precedent should be used within the confines of the limitations of the context established when previously approved. Precedent may be used in whole or in part and should be technically justified. Any similarities or differences should be technically supported and demonstrated as appropriate. Consideration should also be given to any NRC guidance that has been documented from the time of the approval of the original occurrence to the time of the application that uses it as precedent.

1.6 ASSUMPTIONS AND ENGINEERING JUDGMENT

Use of engineering judgment in criticality analyses can result in resource efficiencies. The use of engineering judgment as a basis for an element of the methodology is acceptable as long as the applicant can demonstrate that the rationale behind such determination is sound and can justify that the engineering judgment would not lead to non-conservative results with respect to the regulatory requirements.

The licensee assumptions used in the criticality analysis should be explicitly identified and clearly stated. The licensee should also bear in mind that assumptions can be listed under two categories: explicit and implicit. Explicit assumptions are those the licensee (in this case more specifically the criticality analyst) consciously chooses in preparing the analysis. Implicit assumptions are those the licensee uses that are inherent [i.e., involved in the constitution or essential character of something] to the method. To ensure completeness, and provide clarity to the regulator for the application review, the licensee should clearly identify their assumptions. The licensee, to the extent practicable, should provide a basis supporting assumptions defined in the application. Where no basis exists for the use of engineering judgment, the licensee should modify their approach such that the criticality analyses can be performed without the use of that engineering judgment.

Use of engineering judgment and assumptions may also consist of applying risk insights as part of a “graded” licensing approach and is acceptable as long as the assessments consider relevant safety margins and defense-in-depth attributes. For example, a criticality analysis that demonstrates a maximum k_{eff} with a relatively large margin to the regulatory k_{eff} limit, may be permitted to make more assumptions about results or uncertainties than a criticality analysis that demonstrates a maximum k_{eff} with a relatively small margin to the regulatory k_{eff} limit.

2 ACCEPTANCE CRITERIA

Fresh (New) Fuel Storage

Normally, fresh fuel is stored temporarily in racks in a dry environment (new fuel storage vault) pending transfer into the spent fuel pool and then into the reactor core. However, moderator may be introduced into the vault under abnormal situations, such as flooding or the introduction of foam or water mist (for example, as a result of fire-fighting operations). Foam or mist affects the neutron moderation in the array and can result in a peak in reactivity at low moderator density (called "optimum" moderation). Normal conditions (i.e., dry) need not be addressed in criticality safety analyses since there is no moderator. However, criticality safety analyses must address the following two independent accident conditions with associated limits, which are incorporated into plant technical specifications:

a) With the new fuel storage racks assumed to be loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water, the k_{eff} must not exceed 0.95, at a 95 percent probability at a 95 percent confidence level (10CFR 50.68(b)(2)).

b) With the new fuel storage racks loaded with fresh fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid corresponding to optimum moderation, the k_{eff} must not exceed 0.98, at a 95 percent probability, 95 percent confidence level (10CFR 50.68(b)(3)).

An evaluation need not be performed for the new fuel storage facility for racks flooded with low-density or full-density water if it can be clearly demonstrated that design features and/or administrative controls prevent such flooding.

Under the double-contingency principle, the accident conditions identified above are the principle conditions that require evaluation. The simultaneous occurrence of other independent accident conditions need not be considered.

Spent (Used) Fuel Storage

Criticality safety analyses for pool storage of new and used fuel may utilize one of two available approaches.

1) For pools where no credit for soluble boron is taken (typically BWR pools), the criticality safety analyses must meet the following limit,:

    a. With the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water, the k_{eff} must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, for both normal and accident conditions (10CFR 50.68(b)(4)).

2) For pools where credit for soluble boron is taken (typically PWR pools), the criticality safety analyses must meet two independent limits:

    a. With the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water, the k_{eff} must remain below 1.0

(subcritical), at a 95-percent probability, 95 percent confidence level (10CFR 50.68(b)(4)).

b. With the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity and flooded with borated water, the k_{eff} must not exceed 0.95, at a 95-percent probability, 95-percent confidence level (10CFR 50.68(b)(4)).

3 COMPUTER CODES

3.1 TYPES AND USES OF COMPUTER CODES

A variety of methods may be used for criticality analyses provided the cross-section data and geometric capability of the analytical model accurately represent all important neutronic and geometrical aspects of the storage racks. In spent fuel pool criticality safety analyses there are two general types of computer codes that are used, criticality codes and depletion codes.

The criticality codes are used to determine the eigenvalue (k_{eff}) of the analyzed system. The isotopic concentrations of the materials in the system are determined from manufacturing data and depletion analysis. The Monte Carlo method relies on repeated random sampling to compute the answer. Cross sections are used as probabilities of interaction and the Monte Carlo code then calculates and tracks individual neutron lifecycles. Although many criticality codes utilize Monte Carlo methods, there are other criticality codes that utilize deterministic transport methods.

The depletion codes are used to calculate the nuclide density changes that occur in the fuel during operation in the reactor core. In addition, decay changes in nuclide concentrations due to non-power cooling times are also captured in depletion calculations. In general, depletion codes utilize deterministic transport methods but Monte Carlo methods may also be used.

The codes to perform depletion and criticality calculations rely upon the use of cross-section libraries. Cross-section libraries used in the criticality analysis should be widely accepted and peer reviewed. Cross-section libraries that have previously been found acceptable for use include the multi-group and continuous energy ENDF/B series.

The licensee should state which codes were utilized along with the type/version of cross section libraries. The use of the term computer code in this document means the combination of the computer code and cross-section library. The code version and cross section set used in the analysis should be the same as those used in the validation of the codes.

3.2 COMPUTER CODE VALIDATION

The licensee should describe all computer codes that are used in the criticality safety analysis, including the validation of the codes. Validation of the codes includes benchmarking by the licensee (i.e, the analyst or organization performing the analysis) by comparison with experiments and accounting for the parameters not covered by the existing experiments. This qualifies both the ability of the licensee (analyst/organization) and the computer environment. The critical experiments used for benchmarking should include configurations having neutronic and geometric characteristics comparable to those of the proposed storage facility.

The computer code validation consists of validating both the computer code used in the depletion calculations and the computer code used for calculating the reactivity of the system (i.e., the criticality code). Appendix A contains a discussion of acceptable methods of performing validation of the criticality (Section A.1) and depletion codes (Section A.2).

4 REACTIVITY EFFECTS OF DEPLETION

This section described appropriate methods for performing the depletion analysis for PWR and BWR fuel.

4.1 DEPLETION MODELS

Historially, depletion models consisted of a model to produce one-group cross sections followed by a solution of the isotopic production and loss equations. The one-group cross sections were produced using the flux from an infinite model of pin cells. Although this approach produced good results, modern nodal methods used in core reload design use a two-dimensional lattice model which determines the one group fluxes used in the isotopic production and loss analysis. Separate lattice models are developed for each unique axial plane, such as low enrichment blankets, control rods insertion, and burnable absorbers.

Depletion analysis is performed using nominal geometric dimensions.

4.2 REACTIVITY EFFECTS OF DEPLETION FOR PWRS

The most important parameters that could potentially result in an impact on the reactivity of used fuel in depletion analyses for PWRs are:

a) Relative power during depletion (which impacts the moderator and fuel temperatures during depletion);
b) Soluble boron during depletion;
c) Presence of burnable absorbers; and
d) Rodded operation.

Additional guidance in selecting operating parameters for depletion analysis is provided in NUREG/CR-6665 [17]

4.2.1 Depletion Analysis

Relative Power during Depletion

The relative power of a fuel assembly during depletion will directly impact the moderator and fuel temperature. Higher relative power will result in higher moderator and fuel temperatures. Higher moderator and fuel temperatures during depletion typically result in increased reactivity of used fuel in the storage rack. The moderator and fuel temperature used during the depletion analysis should therefore be conservative and appropriately justified. While higher specific power will lead to a higher Sm-149 content after the decay of Pm-149, which lowers reactivity, this effect is much smaller than the impact of the moderator and fuel temperature. Therefore, the highest relative power should be selected to maximize the net reactivity of all the effects associated with the conditions of depletion.

Since the effects of higher moderator temperature and higher fuel temperature can be approximated as linear [17], it is appropriate to use the maximum burnup-averaged relative power. The burnup averaged relative power is determined by the discharge burnup divided by the sum of the cycle burnups. The maximum burnup averaged relative power would be expected to be a function of the assembly burnup. The analyst may use either a single relative power chosen to bound the relative power over the life of the fuel assembly in the reactor or use a relative power as a function of burnup. Further, the relative power may be a function of fuel management strategy, so the relative power could be a function of cycle fuel management techniques, enrichment, presence of absorbers, etc.

$$\bar{P} = \frac{Assembly\ burnup}{\sum_{i=1} Cycle\ burnup_i} \qquad \text{(Equation 1)}$$

The relative power is an assembly averaged parameter. Many safety analysis calculations use a peak pin power, because the analysis is specific to the effect on individual fuel pins. However, in criticality safety calculations, the reactivity of the system is reliant on the mass of fissile material in more than one fuel assembly. Therefore, it is appropriate to use the relative power for criticality safety calculations based on an assembly averaged parameter.

A conservative moderator temperature would be the core outlet temperature increased by the relative power determined as stated above. A more realistic approach could use the moderator temperature as a function of axial position. If this approach is taken an axial power distribution is required along with justification for its appropriateness. As with relative power, this parameter should be based on assembly average rather than peak channel data.

The fuel temperature used in the depletion calculations is determined based on the burnup averaged relative power. Licensed fuel management tools use models that predict fuel temperature as a function of the linear heat rate and burnup. It is acceptable to use these fuel temperatures and the relative power to determine a conservative fuel temperature (applied either uniformly or as a function of axial height and burnup).

Soluble Boron during Depletion

The soluble boron concentration during depletion can have a significant impact on the reactivity of the fuel in the storage rack. The higher the concentration during depletion, the higher the reactivity of the fuel at a given burnup. It has been shown that treatment of the soluble boron as a burnup averaged value results in the same effect on the fuel reactivity as modeling the actual boron concentration changes as a function of time [30]. A conservatively high burnup averaged soluble boron concentration should therefore be confirmed and used in the depletion calculations.

Burnable Absorbers

PWR reactors use a variety of burnable absorbers during operation for the purposes of reactivity control and power distribution control. These absorbers can be mixed into the fuel pellet (Gd, Erbium, etc.), added as a coating on the fuel pellet (ZrB_2 IFBA) or be included as inserts in the guide tubes (e.g. WABA, BPRA, Pyrex). In all cases the effect of the presence of these absorbers on the reactivity of the fuel assembly should be appropriately considered and accounted for in the depletion analysis. The bounding neutron absorber loading of the burnable absorbers for the maximum burnup should be modeled.

Burnable absorbers harden the energy spectrum during operation due to the presence of the neutron absorber (i.e., absorption of thermal neutrons) and the displacement of water from the guide tubes. The reactivity effect on the fuel assembly is a function of the duration of the removable absorber in the fuel assembly (determined through the amount of burnup the fuel assembly experiences while the burnable absorber is present). Therefore, the maximum burnup that a fuel assembly receives while containing a burnable absorber must be determined and used in the analysis.

Studies have shown that Gadolinium and Erbium burnable absorbers can be conservatively neglected [18]. The residual content of Gadolinium and Erbium and the displacement of fissile material (UO_2) has more negative reactivity worth than the positive worth due to harder spectrum depletion, regardless of the burnup of the fuel assembly. If Gadolinium or Erbium is to be neglected, the volume averaged enrichment may be used in the criticality model. Recent analysis has shown this to be a conservative approach [31].

It is also important to note that multiple absorbers, such as WABAs and IFBAs, can be present in a fuel assembly undergoing depletion in any given cycle. In the event of multiple absorbers, the depletion analysis should take into account all of the burnable absorbers present in the fuel assembly.

Further, other moderator displacing inserts need to be addressed, such as primary and secondary sources. Normally, primary and secondary sources will be covered by the conservatism in the burnable absorber assumptions, but confirmation is necessary.

In all cases the burnable absorbers are modeled with nominal dimensions in the depletion analysis.

Rodded Operation

The criticality safety analysis should include the impact of exposure to fully or partially inserted control rods (and/or part length rods) since rodded operation typically increases the fuel assembly reactivity at a given burnup [19]. Control rod insertion has a similar effect as burnable absorber by affecting the energy spectrum in the core. While most PWRs operate with all rods out (i.e., no partial insertion in the core), use of this assumption should be justified. Separate loading criteria may be developed if different assumptions are used for addressing rodded operation.

Cooling Time

The standard practice is to perform the depletion analysis at a very short cooling time (hours or days) with no Xenon to determine the spent fuel isotopic inventory after discharge. This is commonly referred to a zero cooling time and is intended to represent freshly discharged fuel. However, as the short lived fission products decay and ^{241}Pu decays to ^{241}Am, the fuel assembly reactivity continues to decline. This additional reduction of reactivity with cooling time can be credited to allow for greater flexibility in managing the spent fuel inventory. Many of the modern depletion codes can perform the change in isotopic inventory with additional time automatically.

Other Depletion Parameters

The modeling of down time or part power operation during depletion has been shown to have only a small effect on the assembly reactivity [32]. As discussed above, the use of conservative moderator and fuel temperature based on the highest assembly power for the duration of depletion produces a conservative isotopic concentration.

Flux suppression inserts have been used at a number of plants. Flux suppression inserts are composed of a strong neutron absorber, such as Hafnium, to reduce the flux on the core vessel. Being composed of a neutron absorber, they harden the spectrum and displace water from the guide tubes, similar to the effect associated with control rods and burnable absorbers. Typically, these inserts are placed in fuel assemblies in the periphery of the core, where little additional burnup accumulates while these inserts are present. These inserts require analysis to show that the burnable absorber assumptions cover the reactivity effects associated with flux suppression inserts.

4.2.2 Fuel Assembly Physical Changes with Depletion

During reactor operation, the fuel rods undergo small physical changes. These changes are driven by the behavior of the ceramic uranium dioxide fuel pellets as they generate energy. These may have an impact on the reactivity of the fuel in the SFP environment. The specific physical changes of concern are changes to fuel density, clad outer diameter (OD), and clad thickness. It should be noted data for fuel pellet diameter is also captured because fuel pellet diameter changes are directly correlated to fuel density changes.

The impact of fuel geometry changes on reactivity changes was analyzed in a proprietary Westinghouse study which is summarized here. The study is based on calculations performed with NRC-approved fuel performance and fuel depletion codes [38,39] for a Westinghouse 3-loop PWR core operating with a 15x15 fuel lattice. The study included both IFBA and non-IFBA fuel, modeled fuel pellets near both the center and top of the assembly, and covered a burnup range from 0 – 62 GWD/MTU. The study was divided into three major sections:

1. Modeling the physical behavior of fuel during operation using the PAD code to determine the minimum and maximum values for fuel density, clad OD, and clad thickness;
2. Modeling the depletion of the fuel with the PARAGON using the minimum and maximum values calculated with PAD to determine fuel assembly isotopics; and
3. Determining the changes in reactivity due to the physical changes in the fuel over depletion.

The change in the physical behavior of the fuel during operation is provided in Figure 4-1 through Figure 4-4. These figures are based on PAD/PARAGON calculations and represent plant-specific values, however, their importance is in the demonstration of the behavior of fuel rods with irradiation. The behaviors exhibited by the pellet and clad are not specific to any reactor design, they are applicable to all UO_2 fueled plants. Therefore, the values on the y-axis are omitted because their inclusion could lead to focus on specific values rather than the general behaviors generated by depletion.

Figures 4-1 and 4-2 show the density and diameter changes of the fuel pellet with respect to depletion. Figure 4-1 shows that the pellet density initially increases to a maximum very quickly and then decreases with additional depletion. Figure 4-2 shows the corresponding pellet diameter changes with depletion. Both figures clearly demonstrate the two widely known phenomena of fuel densification and fuel swelling. Early in reactor operation the heat generated by fission causes fuel to densify and the fuel pellet diameter to correspondingly decrease. As operation continues, the fission products produced in the pellet cause the pellet diameter to expand and the fuel density to decrease. It should be noted that while the fuel density is changing, this is solely due to changes in pellet dimensions as the mass within the fuel is unchanged.

Figures 4-3 and 4-4 show the changes in cladding thickness and outer diameter due to irradiation. The behavior of these parameters align with the behavior of the fuel pellets. Initially the clad OD decreases, thickening slightly, until the clad comes into contact with the fuel pellet. Once the clad and pellet come into contact, the clad expands and thins as the fuel pellet swells and causes the rod diameter to expand.

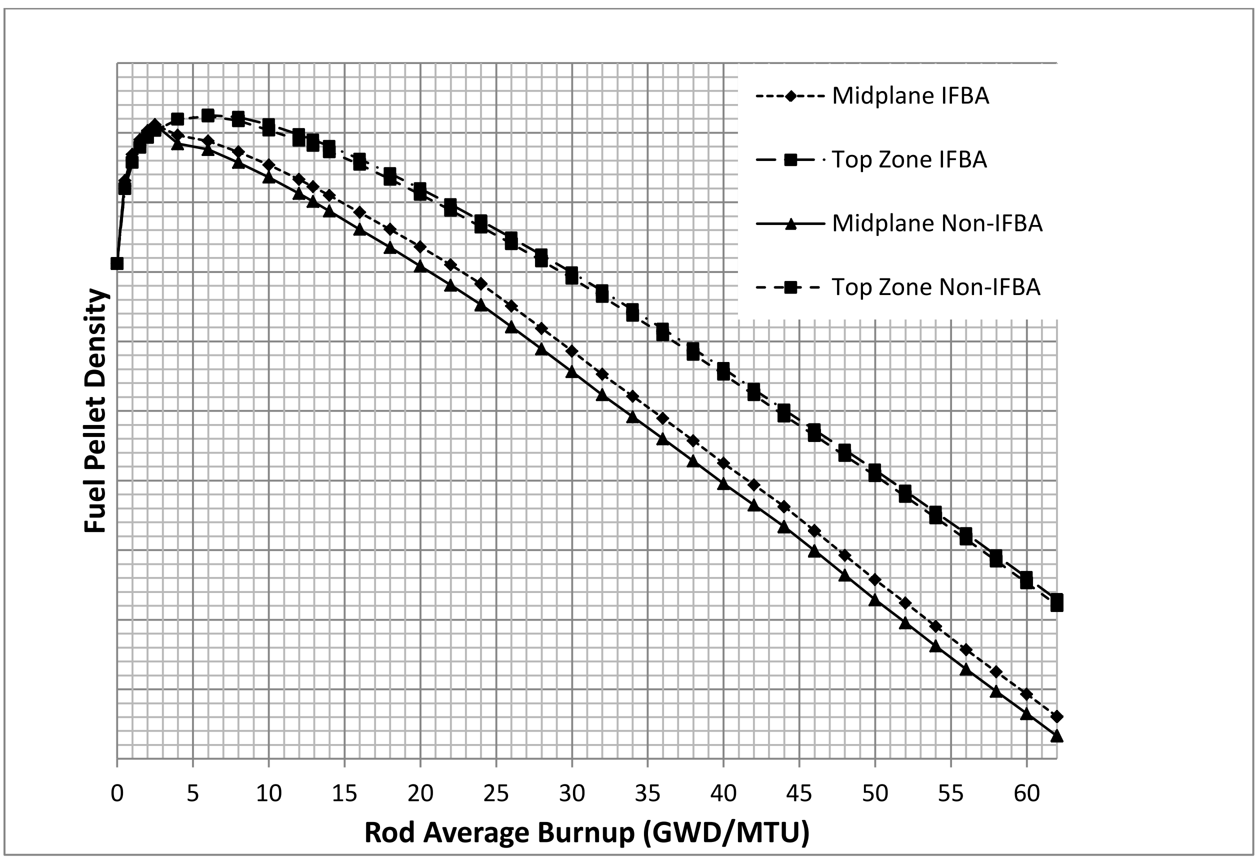

Figure 4-1: Fuel Density Behavior over Depletion

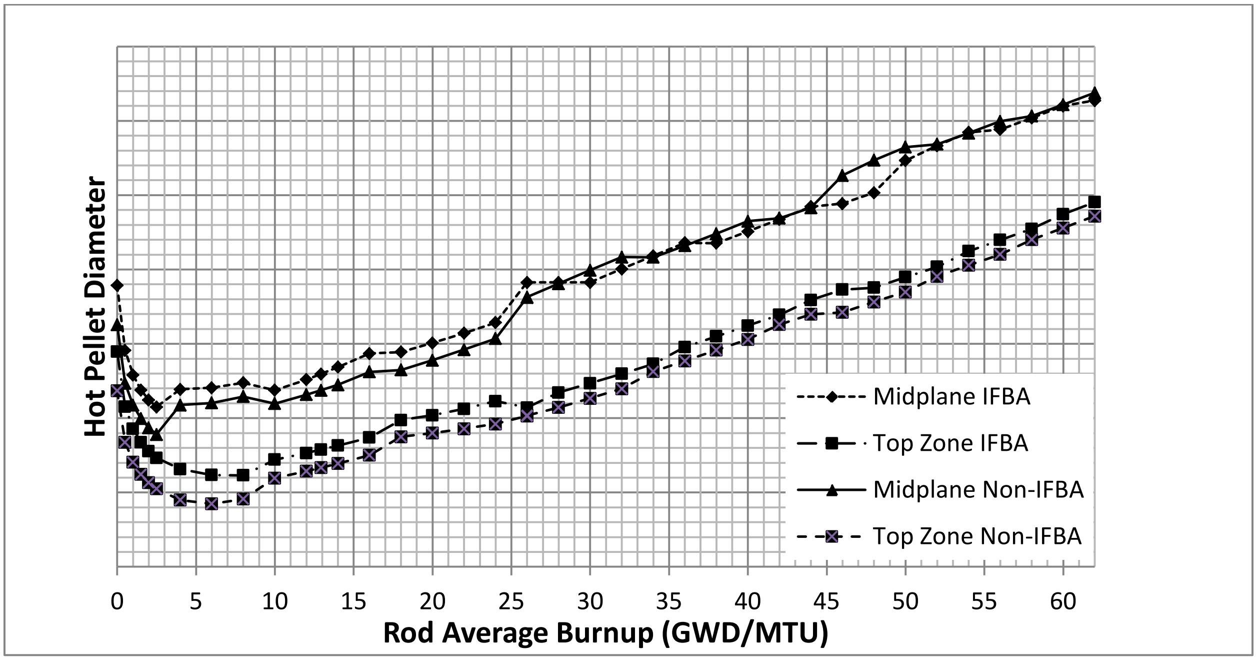

Figure 4-2: Fuel Pellet Diameter Behavior over Depletion

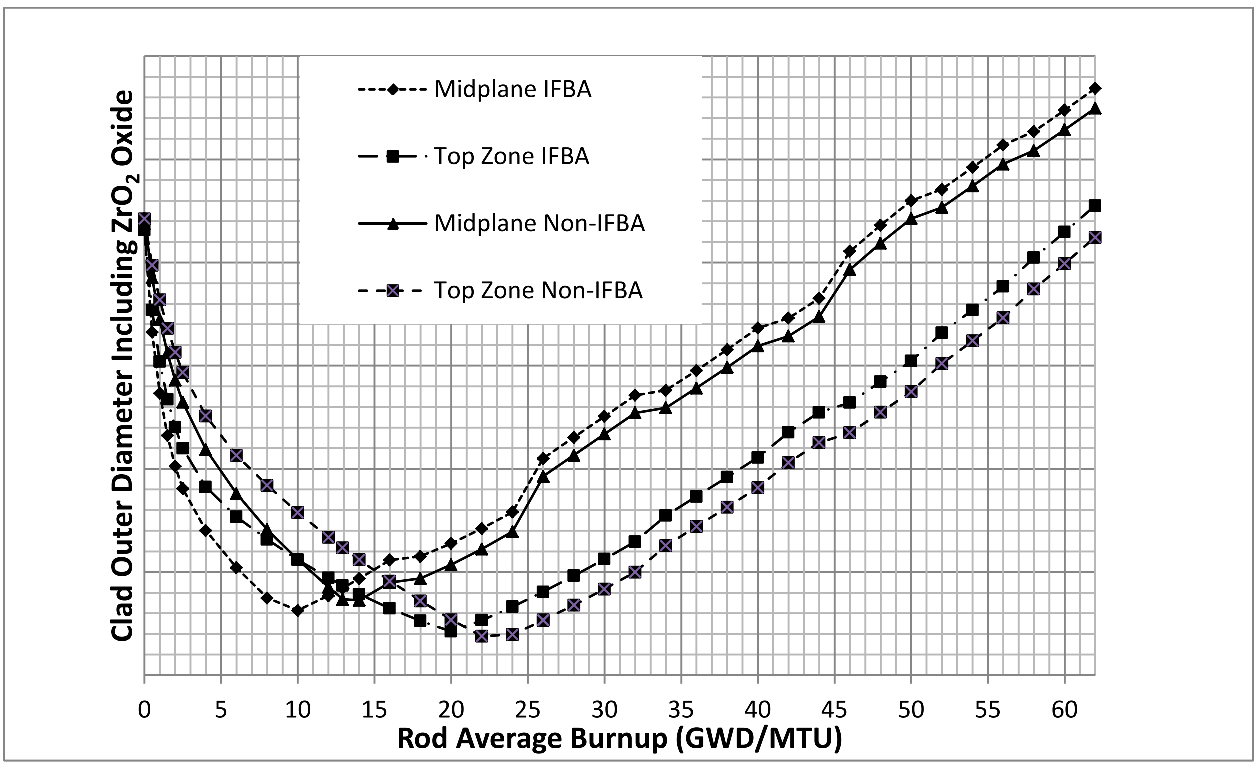

Figure 4-3: Clad Diameter Behavior over Depletion

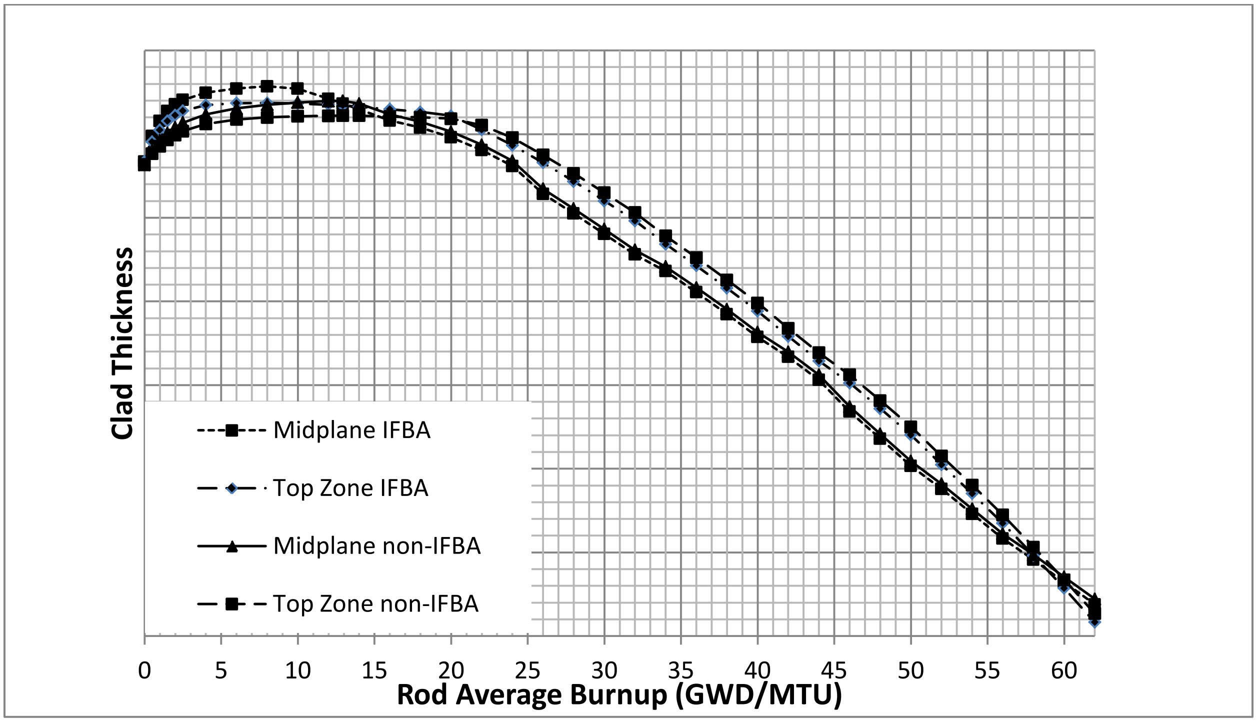

Figure 4-4: Clad Thickness Behavior over Depletion

Based on the pellet and clad data developed above, depletion and reactivity calculations were performed with PARAGON and KENO V.A respectively. These calculations used the minimum and maximum pellet and clad data points to develop conservative estimates of the reactivity impact of the fuel changes during depletion. The depletion calculations assumed either the maximum or minimum value for the parameter in question throughout depletion. The parameters are each treated individually in determining the reactivity impact, although it should be noted that the fuel density and fuel pellet diameter are treated together because they are different aspects of the same parameter.

The results of the reactivity calculations indicate that certain changes in fuel geometry causes an increase in reactivity when looked at in isolation. However, there are also individual fuel geometry changes which cause a decrease in reactivity. Because none of these parameters are truly independent of the others, an additional set of cases was performed incorporating all of the changes associated with fuel depletion together. This calculation provides a more accurate assessment of the actual neutronic importance of these changes.

To provide a better estimate of the true reactivity impact, two calculations for both IFBA and non-IFBA fuel were performed. The first condition evaluated the fuel pin geometry associated with peak fuel density and the second condition evaluated end of life conditions (62 GWD/MTU). The results of these calculations are provided in Table 4-1.

Table 4-1: Overall Reactivity Impact of Fuel Changes	
Case Name	**Δk**
non-IFBA End of Life	-0.00093
non-IFBA Maximum Density	-0.00123
IFBA End of Life	-0.00040
IFBA Maximum Density	-0.00409

The results of Table 4-1 demonstrate that each individual fuel geometry change has an impact on fuel reactivity. When these changes are looked at holistically, the overall impact of fuel geometry changes with depletion is small. These results are not unexpected because it aligns with standard procedures for performing fuel management calculations. These procedures essentially ignore fuel geometry changes, which would not be the case if they had a significant role (either positively or negatively) on fuel reactivity. Based on this study and its alignment with general fuel management practices, fuel geometry changes with depletion do not need to be explicitly modeled in depletion calculations.

4.2.3 PWR Depletion Uncertainty

Historically, engineering judgment was used to estimate the uncertainty associated with fuel depletion calculations as a percentage of the change in reactivity associated with depletion [24]. Two independent evaluations have been conducted to determine the magnitude of this uncertainty and show that the use of 5% Δk as an uncertainty is conservative for the cross-section sets ENDF/B-V through ENDF/B-VII. [27, 33, 34] Both analyses also confirm that a zero bias is appropriate. When calculating the depletion uncertainty, the change in reactivity between the zero burnup, fresh fuel condition and the burnup of interest is determined without burnable absorbers.

For low burnup, the depletion reactivity benchmarks suggest a higher uncertainty, however this is based on limited data at low burnups. The chemical assay approach, using the direct difference method, results in an uncertainty that is less than 5% even at low burnups [34]. Therefore, a simple approach of using 5% of the change in reactivity associated with depletion is an acceptable method for accounting for the uncertainty associated with the depletion calculations for all burnups. Because these methods are an integral benchmark of the entire system modeled by the depletion codes it covers all uncertainties associated with depletion, such as uncertainty in computation of the isotopic inventory by the depletion code, uncertainty in cross-sections (both actinides and fission products), etc. Therefore, no other uncertainties are needed to be applied to the calculation of the maximum k_{eff}.

Licensees may use of the 5% based on the justification provided above. However, if credit is desired for a more accurate assessment of the depletion uncertainty, a licensee may use either of the approaches above and detailed in Appendix A [27, 33] to reduce the conservatism in this parameter.

4.3 PEAK REACTIVITY ANALYSIS FOR BWRS

It is standard practice that BWR spent fuel pool criticality analyses are performed at the burnup that produces the lattice peak reactivity. BWR fuel lattices that contain an integral burnable absorber typically result in a lattice peak reactivity at a specific burnup value, usually under 25

GWD/MTU, due to the positive reactivity from the depletion of the integral burnable absorber competing with the negative reactivity from the depletion of the fissile material.

The general methodology for BWR spent fuel pool criticality analyses is to perform in-core depletion calculation for the various assembly designs in use, then to restart the calculations with the assemblies in the standard cold core geometry (SCCG) and then in the storage rack geometry. The SCCG is defined as an infinite array of fuel assemblies on a 6-inch lattice spacing at 20ºC, without any control rods or voids. The burnup at the limiting k_{inf} in the SCCG is determined and then the k_{inf} in the storage rack geometry is calculated at this burnup. A reactivity allowance for applicable biases and uncertainties is added to the calculated k_{inf} in the rack geometry and the resulting k_{eff} is compared to the regulatory limit of 0.95.

A licensee should perform calculations in a manner that accounts for both the radial and axial pin locations. The peak reactivity method inherently bounds all axial effects by modeling the peak axial reactivity across all exposures at all axial nodes.

4.3.1 Depletion Parameters

A licensee should account for the dependence of the peak reactivity burnup and the magnitude of the peak reactivity for all storage rack calculations that are used to determine the maximum in-rack k_{eff} in the analysis. The following parameters can have a significant impact on reactivity in the storage rack and therefore should be considered:

- **Reactor operating parameters**:
    - Void fraction – Higher void fractions typically increases peak reactivity, however, this is dependent upon the other reactor operating parameters and the full range of void fractions should be considered in conjunction with the other reactor parameters.
    - Moderator temperature – The moderator temperature is typically a fixed value in a BWR and should be considered in conjunction with the values appropriate to the reactor operation at power. Note that higher moderator temperatures typically result in an increase in peak reactivity in the storage racks.
    - Fuel temperature – Higher fuel temperatures typically results in an increase in peak reactivity in the storage racks.
    - Power density – The power density typically has a lower impact on peak reactivity than the other reactor parameters and the value used can be chosen based on its relationship to the fuel temperature.
- **Non-reactor operating parameters**:
    - Lattice specific parameters. Lattice specific parameters should each be evaluated during depletion and in the storage rack for their impact on peak reactivity. These parameters should at a minimum include:
        - Number, location and concentration of integral burnable absorber fuel rods
        - Number and location of partial length rods

A licensee should consider the following when preparing the depletion analysis for a submittal of a license application:

- All BWR criticality calculations should ensure a conservative reactivity is analyzed in the storage configuration with consideration given to possible cooling and discharge times.
- The reactivity effects of the reactor operating parameters can be applied either as separate biases or included in the design basis models. When limiting reactor operating parameters are included in the design basis models, the analysis should determine and use the combination of reactor operating parameters that result in the bounding peak reactivity in the SFP rack geometry and all calculations that are used to determine the maximum in-rack k_{eff}, including non-reactor parameter studies. Due to the large variation of BWR fuel designs and lattices within designs, the bounding reactor operating parameters may not be applicable to another design or lattices and therefore further evaluations may be needed to show which parameters are bounding for other fuel designs or lattices within a design. The non-reactor operating parameter studies may demonstrate a peak reactivity burnup and a peak reactivity magnitude that varies from the design basis model and should be accounted for in the analysis by appropriate inclusion of the magnitude of the reactivity difference due to the change in peak reactivity.

4.3.2 BWR Depletion uncertainty

The BWR lattice physics/depletion codes used for SFP criticality analyses are the same depletion codes used and validated for BWR core design and core monitoring applications. In these applications, the integral burnable neutron absorber burnout is very important, so there is high confidence that the integral burnable neutron absorber depletion is accurate within 5%. It is additionally noted that 5 percent of the reactivity decrement to burnup of interest is reasonable for BWRs given that PWR depletion uncertainty validation with measured power flux data has demonstrated the 5 percent of the reactivity decrement is conservative and they are very similar, both being thermal, light water reactors with low enriched UO_2 fuel.

The reactivity decrement to the burnup of interest is, specifically, the cold, beginning-of life (BOL) reactivity of the spent fuel rack analyzed bundle with no integral burnable neutron absorber present compared to the reactivity of the cold, analyzed bundle at the exposure statepoint used in the analysis as shown in Figure 4-5. Both reactivities are calculated for comparison in the rack system. Five percent of the difference in reactivities between these two cases is included as an uncertainty to the spent fuel pool criticality analysis to cover the depletion isotopic benchmarking gap. Figure 4-5 shows how to determine the reactivity decrement for BWR criticality analysis where the burnup of interest is the peak reactivity.

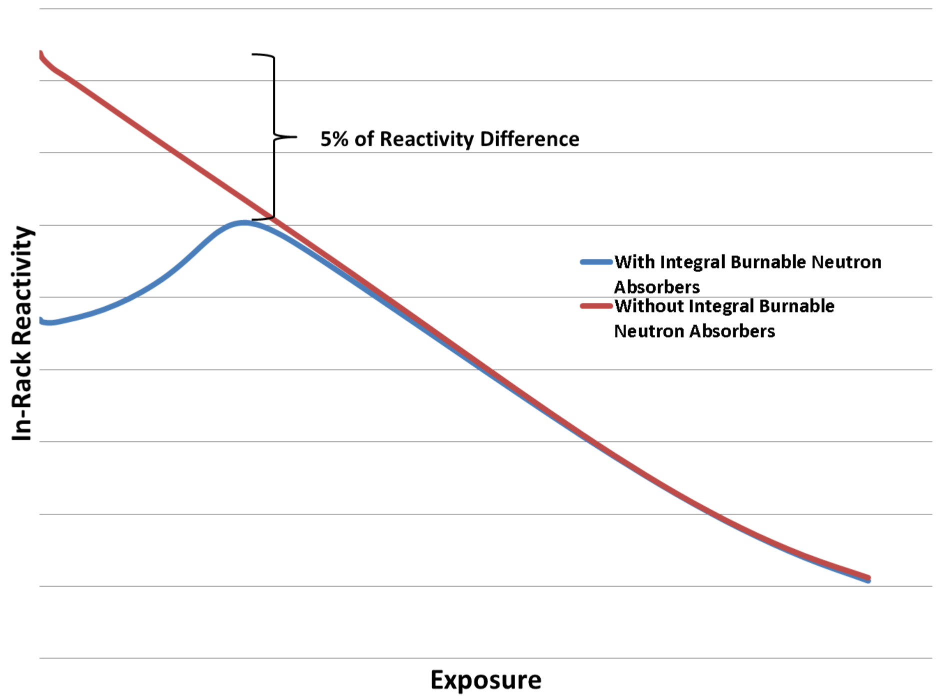

Figure 4-5: BWR Peak Reactivity Depletion Uncertainty

The depletion uncertainty covers the uncertainty in the change in macroscopic cross-sections (i.e., the change in isotopic number densities generated during the depletion simulations multiplied by the microscopic cross-sections). If a reduction in the recommended 5% of the delta k of depletion is required and a chemical assay approach is used, the uncertainty from the minor actinides and fission product cross sections, which are not explicitly represented in the critical experiments needs to be covered. This can be done by adding a bias of 1.5% of the worth of the minor actinides and fission products. This approach has been developed and presented in NUREG/CR-7109.

5 RACK AND FRESH FUEL MODELING

5.1 FUEL ASSEMBLY MODELING

5.1.1 Design Basis Fuel Assembly

Most, if not all, spent fuel pools contain several different designs of fuel assemblies. In the case of PWR pools, this is typically limited to two or three different designs that are geometrically very similar with only minor changes that have a relatively small effect on reactivity (grid spacer, mixing vane modifications). BWR pools, however, typically have many more fuel assembly designs with significant geometric differences (i.e., different array sizes, differences in the number, location and shape of water rods, presence of partial length fuel rods). Regardless of the differences, it is convenient to establish a single fuel assembly design to be used in all depletion and criticality calculations for simplicity and consistency.

Calculations need to be performed for each unique rack design and fuel assembly type, using nominal dimensions to establish which fuel assembly type is most limiting. It is also important to address the change in reactivity with depletion, as the bounding fuel type can change with burnup (because of differences in the fuel to moderator ratio between different fuel designs, a fuel assembly that is bounding at fresh fuel conditions, may not be limiting at other burnups). The design basis fuel assembly is that assembly that provides the most limiting reactivity at a given burnup and enrichment. In the case where a single fuel assembly is not bounding over all burnup and enrichment combinations, the difference between the design basis assembly and the other more bounding assembly type(s) is applied as a bias to the calculation of maximum k_{eff}.

In the determination of the design basis assembly it is acceptable to use a hybrid set of parameters from multiple assemblies that result in a bounding, more limiting design basis assembly. A prime example of this approach is the use of the maximum nominal fuel density that bounds all fuel designs in the spent fuel pool. This approach also provides additional conservatism in the analysis.

When significant differences occur between designs, it is acceptable to have more than one design basis fuel assembly.

5.1.2 Fuel Assembly Manufacturing Tolerances

As described in Section 5.1, criticality analyses rely on a nominal representation of the fuel assembly design (i.e., nominal dimensions, materials, and isotopic concentrations). However, each individual parameter is manufactured within specified tolerances to ensure quality control, fabricability, etc.

The following fuel assembly tolerances should be considered for inclusion as uncertainties in the criticality analysis, unless they can be shown to be insignificant:

a) Enrichment
b) Channel (BWR only)
c) Pellet Density
d) Rod Pitch
e) Fuel Pellet Outside Diameter
f) Cladding Outside Diameter

If independent uncertainty items are evaluated separately, the total k_{eff} uncertainty is the root sum square (RSS) of the individual k_{eff} uncertainty values. Alternatively, the analysis could calculate k_{eff} with all tolerance values selected to maximize k_{eff}. It is also acceptable to use a combination of these two approaches. For example, a maximum pellet density may be used and the other parameters are statistically combined.

To ensure that the maximum reactivity is being calculated per the requirement of 10CFR50.68 [1], effects of tolerances should be considered for each parameter that may contribute to a significant positive reactivity effect. Significance is determined based upon the overall effect on the total uncertainty, and on the margin to the regulatory limit. Because the total uncertainty term is typically dominated by a few large uncertainties, an individual uncertainty that is less than

10% of the total uncertainty may be considered insignificant. For example, suppose the total uncertainty (defined to be the square root of the sum of the squares of independent uncertainties or RSS) is 0.01 Δk. Using RSS, the effect of an additional independent uncertainty equal to 10% of the total uncertainty (0.001 Δk) can be calculated to increase the total uncertainty from 0.01Δk to only 0.01005 Δk. Unless the margin to the regulatory limit is very small, the 0.001 Δk uncertainty is not significant compared to the total uncertainty.

The significance of some uncertainty values may vary with storage conditions (e.g. soluble boron and rack design). Fuel assembly tolerances should be evaluated in the appropriate rack model. The criticality analysis should demonstrate that the uncertainty values used are appropriate to the storage conditions by using either condition-specific values, bounding values, or application of additional k_{eff} margin to the regulatory limit.

Tolerances on the fuel clad thickness and guide and instrument tube thickness have been shown in a generic study to be insignificant and do not require analysis [31]. The clad thickness tolerance is insignificant since zirconium has a small absorption cross section. Since the inside of the clad is a gap filled with helium, the substitution of gas for zirconium has very little reactivity effect. While changing the guide and instrumentation tube thickness does affect the amount of water, the number of guide/instrument tubes is less than 10% of the number of rods in an assembly and this low volume makes the reactivity of the tolerance negligible.

5.1.3 Axial Burnup Distribution

When modeling the fuel assembly in the criticality analysis, the reactivity is affected by the distribution of burnup along the axial length of the fuel assembly. The burnup distribution is affected by the operating conditions (temperatures, flux, presence of inserts, etc). The axial burnup distribution starts out generally cosine shaped and gradually flattens in the middle of the assembly. Additionally, the the neutron flux and power shifts to the ends of the assembly at the end of the fuel assembly life in the reactor. The lower burnup near the ends of the assembly combined with the lower moderator density at the top of the assembly, causes the region near the top of the fuel assembly to control the reactivity of the entire assembly. Therefore, the nuclear criticality analysis should consider an appropriate representation and nodalization of the burnup profile that encompasses a bounding shape of the licensee's inventory. Three options are provided for licensees to choose from for modeling of the axial burnup distribution, depending on the amount of information available to support the analysis and the level of verification for future fuel assemblies to meet the axial burnup distribution used in the analysis. In all options, the results with an explicit axial burnup distribution should be compared to the axially uniform profile, which assumes the same burnup along the entire axial length.

Option 1: Use of Generic Axial Burnup Distributions from NUREG/CR-6801

NUREG/CR-6801 [20] evaluated 3169 axial burnup profiles to determine the most reactive representatives in each burnup range. Included in the population are B&W 15x15, CE 14x14, CE 16x16, Westinghouse 15x15 and Westinghouse 17x17 profiles. The profiles in the database include fuel designs that contain burnable absorbers that have been and continue to be used, including borosilicate glass, zirconium diboride (IFBA), WABAs, Gadolinium and Erbium. Additionally, the profiles include assemblies exposed to control rod insertion, including axial power shaping rods (APSRs). Given the broad range and applicability of the database, along with the selection of the axial burnup profile in each burnup range that produces the limiting reactivity, it is appropriate and conservative to use the NUREG-/CR-6801 profiles for PWR reactors.

The database does not include axial burnup profiles associated with fuel assemblies containing lower enriched axial blankets at the top/bottom of the fuel assembly. However, as stated in NUREG/CR-6801,

> *"because the axial blankets have significantly lower enrichment than the central region, the end effect for assemblies with axial blankets is typically very small or negative... consequently, profiles from assemblies with axial blankets were not considered…"*

It is acceptable to use the profiles from NUREG/CR-6801 to bound axially blanketed fuel assemblies. It should be noted that this does not allow for credit of the lower enrichment of the axial blanket region in the criticality analysis.

Because of the broad range of applicability and the conservative nature of using the most reactive axial burnup profile for each identified burnup range, there is reasonable assurance that axial burnup profiles from future discharged fuel assemblies will also be bounded by the database of profiles contained in NUREG/CR-6801. If drastic changes are made to the core operation (i.e., load following, gray rods, etc.), it should be verified that the new axial burnup distributions still behave in a similar manner as to before the core design change.

The NUREG/CR-6801 limiting shapes were selected assuming the rack is uniform axially. If the rack has reduced length absorber panels that leave a significant portion of the active fuel outside of the absorber panels, new limiting axial burnup distributions must be determined.

Option 2: Use of Plant Specific Bounding Profile(s)

Core management tools and advanced nodal codes have the ability to calculate the axial burnup distribution for each fuel assembly as a function of burnup throughout the cycle of operation. These axial burnup distributions are used to ensure the core operates within the limits specified for the reactor. These axial burnup profiles can also be used in the spent fuel pool criticality calculations. One conservative approach is to take the plant-specific population of axial burnup distributions and determine a bounding axial burnup profile specific to the fuel assemblies being stored in the spent fuel pool. A simple approach to create this bounding axial burnup profile is to take the minimum relative burnup of each node (there are typically between 10 and 25 nodes along the entire axial length) from all assemblies on-site at the specific licensee plant. To ensure that the composite axial burnup profile is conservative, no renormalization is performed. This typically provides a weighted relative burnup between 0.95 and 0.98.

Option 3: Use of Most Reactive Plant Specific Profile (s)

The third option also uses plant specific axial burnup profiles through the use of the most limiting profile(s) from the current population of fuel assemblies at the site. This approach involves determining which profile(s) are limiting, such as identifying those profiles with the lowest relative burnup in the nodes closest to the ends of the assembly. This approach ensures that all past discharged fuel is bounded and provides a level of reasonable assurance that future profiles will also be bounded, provided the reactor is operated in a similar manner (e.g., no increase in rodded operation, or new burnable absorber materials are introduced). However, it is recommended that the axial burnup distribution of future fuel assemblies continue to be verified to be bounded by the limiting axial burnup profile(s) used in the analysis using the same method as was used to determine the most limiting profile. This verification would be controlled by the licensee through administrative procedures.

Nodalization

The number and size of the nodes in the axial burnup distribution are an important consideration in ensuring the effect of the low burnup ends of the assembly are properly modeled. Previous studies have investigated the sensitivity of k_{eff} to the nodalization structure of the axial burnup distribution. NUREG/CR-6801, Appendix A [20] concludes:

> *"Results of variations in the size of axial zones in fuel assembly models indicated that for the most part,use of 18 uniform-height axial zones is sufficient to capture burnup distribution effects"*

Additionally, ORNL/TM-1999/99 [36] also found burnup distributions with even fewer nodes to be sufficient under the following circumstances:

> *"Calculations with as few as seven axial zones (three 1/18th-length zones at either end and one large central zone) were shown to converge to the same solution as an 18-uniform-zone model."*

These two references are consistent in recognizing the importance of the size of the nodes at the ends of the assembly (approximately 8 inches or less) and the non-importance of the nodal structure at the center of an assembly modeled with a distributed axial burnup profile. Therefore, the analyst should confirm that the nodes of the axial burnup distribution are appropriately sized, especially at the ends of the assembly.

5.1.4 Reactor Record Burnup Uncertainty

The reactor record burnup uncertainty, also referred to as burnup measurement uncertainty, (BMU) is an uncertainty representing the maximum potential reactivity impact of deviations between an assembly's "true" burnup and the burnup based on reactor records. There are a number of ways to calculate BMU, with each method assuming some value which represents the percent deviation between true and reactor record burnup. This value is typically assumed to be 5% and the effect is statistically combined with other uncertainties. Alternatively, utilities can reduce burnup of assemblies by 5% instead of incorporating the uncertainty. Reducing the burnup of each assembly is effectively the same as treating the BMU as a bias instead of an uncertainty.

Both EPRI and ORNL have performed studies to evaluate the accuracy of reactor records [21, 35]. The EPRI and ORNL reports agree that burnup estimations based on the flux measurements followed by time integration are within 5% of the true assembly burnup, and as such using 5% as the BMU is conservative. It should be noted that both studies indicate that when using properly calibrated core follow software which is updated with in-core measurements the uncertainty is less than 2%.

Therefore, the burnup uncertainty should be accounted for either by including a stand-alone uncertainty in the calculation of the sum of biases and uncertainties, or directly reducing the burnup of assemblies before storing them in the SFP.

5.1.5 Assembly Inserts

In addition to the modeling of the fuel assembly as described above, in some cases the burnable absorber inserts contained in the fuel assembly are also modeled and/or credited in the criticality analysis. This is separate from the effects of these devices during depletion as described in Section 4.2.1.

Control rods are considered "used" when they meet their mechanical or nuclear design limits. This occurs before there is any significant reduction in their neutron absorbing properties for most of the control rod blade. These used control rods can be credited in the criticality analysis to hold down reactivity in assemblies and allow lower burnup requirements. Although neutron absorbing properties are not significantly diminished for used control rods, a conservative reduction should be considered based on the in-reactor usage of control rods.

Non-irradiated removable burnable absorbers (i.e., WABA's, BPRA's) can also be credited to provide additional reduction in the required burnup for storage. The primary effect is associated with moderator displacement from the guide tube and can provide some small benefit.

Fresh fuel often has fuel rods containing burnable absorbers inside the clad as a pellet coating or mixed in with the fuel (i.e. IFBA). The absorbing properties of these burnable absorbers are allowed to be credited in unirradiated fuel.

5.2 STORAGE RACK MODELING

5.2.1 New Fuel Vault

While the New Fuel Vault is a dry environment for unirradiated fuel assemblies, both full (100% density) moderator condition as well as optimum low density moderator condition (i.e., mist or foam) should be considered to ensure the maximum reactivity condition is represented, per 10CFR50.68 [1] requirements.

Usually, the storage racks in the new fuel vault are designed with large lattice spacing sufficient to maintain a low reactivity under the accident condition of flooding. Specific calculations, however, are necessary to assure the maximum k_{eff} is no greater than the regulatory limits. In the evaluation of the new fuel vaults, fuel assembly and rack characteristics upon which sub criticality depends should be explicitly identified and evaluated.

The following vault tolerances should be, at a minimum, considered when evaluating the uncertainties:

a) Cell/Storage Location Pitch
b) Storage Cell Wall Thickness (if present)

Tolerance calculations should be performed for both moderator conditions (i.e., full and optimum).

5.2.2 Spent Fuel Pool Racks

The spent fuel pool rack criticality model consists of a representation of the dimensions and materials of construction, including any installed neutron absorber as well as flux traps (if present). To ensure the model captures any reactivity increases due to uncertainties associated with manufacturing tolerances, each parameter that may contribute to a significant positive reactivity effect should be evaluated. The following spent fuel pool rack tolerances should be, at a minimum, considered when evaluating the uncertainties due to tolerances:

a) Flux Trap Size
b) Cell/Storage Location Pitch
c) Eccentric Fuel Positioning (in racks without absorber panels)
d) Storage Cell Wall Thickness

5.2.2.1 Spent Fuel Pool Temperature

The spent fuel pool temperature affects the reactivity of the storage racks through changes in the cross-sections (i.e., Doppler broadening and changes in the moderator density). The criticality analysis should include calculations at the maximum water density (4^o C) and the maximum temperature allowed for normal operation. The temperature producing the maximum reactivity should be used when comparing against the acceptance criteria. Generic analysis has been performed to show that the most limiting condition will always be found by analyzing the highest and lowest temperature allowed [31].

5.2.2.2 Dimensions

The modeling of SFP rack dimensions is described in Section 4.3. Fixed neutron absorbers are typically part of the original rack design. Rack manufacturer drawings will provide detailed dimensions for the neutron absorber including how the absorber is held in place.

For neutron absorbers that are installed after the original rack construction (i.e., rack inserts), the dimensions are also provided by the manufacturer through drawings or design specifications. The modeling of these absorbers should be consistent with these dimensions and with how they are installed in the SFP.

Manufacturing dimensional tolerances of the neutron absorbers should be included in the uncertainty analysis. Tolerances for absorber length (if shorter than active fuel length), width and thickness should be considered in the analysis. Minimum values for the length and width may be used in lieu of tolerance analyses.

Many racks have thin stainless steel sheets covering the neutron absorber material. The reactivity effect of the manufacturing tolerance on this cover is negligible in non-flux-trap rack designs that contain absorber panels and further calculations for this reactivity is not required [31]. For flux-trap rack designs, the uncertainty due to the manufacturing tolerance on these sheets is small, but cannot generically be declared negligible. Similarly, in racks without neutron absorber, the steel of the sheathing has a small non-negligible effect.

5.2.2.3 Rack Neutron Absorbers

In order to increase the capacity of SFPs, many utilities performed re-racks with high density spent fuel racks. These racks incorporated neutron absorbers (typically containing boron) into the design to allow for higher density fuel storage. Additional absorbing capability may be added to the racks through the use of neutron absorbing rack inserts. The criticality analysis should include a detailed model of these neutron absorbers in order to ensure that they are effective in their intended function to prevent criticality in the SFP. Criticality analyses involving neutron absorber materials include modeling of the boron content (^{10}B areal density) and dimensions. Of these modeling parameters, ^{10}B areal density has by far the largest effect on k_{eff} (as compared with neutron absorber dimensions and non-neutron absorbing material compositions).

There are many different neutron absorbers in use in SFPs. For a detailed description of different neutron absorber materials, see the Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications [29].

Typically, neutron absorbers are not used in dry new fuel vaults, where the geometry of the vault is designed to prevent criticality.

5.2.2.3.1 Boron Content

The boron content of the neutron absorber (B-10 areal density) is an important parameter in the SFP criticality analysis. A conservative approach to modeling the boron content is to assume the minimum boron concentration (typically described in terms of areal density in g/cm^2 ^{10}B) for every neutron absorber panel. This is conservative because all panels actaully placed in service have higher boron concentrations, since the manufacturer must take into account manufacturing tolerances. For example, the manufacturer will target a nominal boron concentration that they can assure an acceptable minimum concentration accounting for manufacturing tolerances. In addition, the manufacturer will fabricate to an as-built minimum that is higher than the certified minimum to further account for manufacturing tolerances.

One approach is to use the minimum as-built areal density that is documented in the manufacturing records. The minimum as-built areal density is the lowest boron concentration measured from all of the panels. Thus all panels actually placed in service have boron concentrations at or above this minimum concentration, and these are documented in QA records. In some cases, these records have been collected by the manufacturer and provided with delivery on a batch basis.

An alternative approach is to use the minimum certified areal density. This is based on the material purchase specification, and the manufacturing process must confirm that the boron content of all panels are above the minimum certified areal density in order to be acceptable for use. The minimum certified areal density is typically less than, and never greater than, the as-built minimum areal density, since QA records will document that all panels have boron concentrations at or above the minimum certified areal density. These QA records are verified prior to storing fuel in the spent fuel pool or new fuel vault racks.

5.2.2.3.2 Neutron Absorber Aging Effects

Certain neutron absorbers may undergo aging effects (i.e. changes in material dimensions or composition over the service life of the neutron absorber). The mechanisms for undergoing changes and the potential impact on their ability to perform their criticality control function are typically specific to the absorber material and rack design. The criticality analysis should clearly identify the absorber assumptions and inputs. If material changes are anticipated over their intended service life, these anticipated changes should be appropriately bounded by the criticality analysis. In extreme cases, if degradation, loss of ^{10}B areal density is anticipated, then appropriate margin to account for the degradation should be included in the criticality analysis sufficient to ensure the analysis is bounding for the intended service life of the pool.

Neutron absorber performance and aging characteristics are monitored through a monitoring program (see Section 9.5). If any un-anticipated aging or change is identified through the monitoring program, then it should be evaluated to determine if there is any impact on the criticality analysis and whether other licensee programs should be utilized (e.g., 10 CFR 50.59 [8] process, operability evaluation).

6 CONFIGURATION MODELING

6.1 NORMAL CONDITIONS

The criticality analysis should consider normal conditions and operations that occur in the spent fuel pool. It is not sufficient to consider only the static condition where all fuel assemblies are in the approved storage locations. It is just as important to consider normal activities and operations in the spent fuel pool. Examples of these normal activities are movement of fuel in and around the spent fuel pool, fuel inspection and reconstitution. Normally the limiting condition is the static condition. Fuel inspections and reconstitution operations are generally separated from the rest of the pool by empty cells. Although the criticality analysis should consider normal conditions, generally calculations are only required for the static condition. Each different normal condition at a plant should be evaluated and if it is potentially more limiting than the static condition, then it should either be considered as a potential starting point for accidents or restricted to make it less limiting than static storage. It is noted that different plants will have different normal conditions.

6.2 INTERFACES

In the event the spent fuel pool contains more than a single storage configuration or storage rack design, the criticality analysis should consider the interface between adjacent storage configurations. An interface occurs every time two or more different storage configurations can be adjacent to one another. In some cases, interfaces may result in a higher k_{eff} than the k_{eff}'s of the configurations evaluated individually. If necessary, interface restrictions may need to be applied to provide conditions for certain storage configurations to be placed next to one another.

Previous guidance provided in DSS-ISG-2010-01 [25] has provided two possible paths to show acceptability of storage configuration interfaces. The first option was to use the maximum biases and uncertainties from the individual storage configurations. The second option was to determine biases and uncertainties specific to the interface. This first option will always show a reactivity higher than either of the two storage configurations that are part of the interface. This could lead to the conclusion that there is an increase in reactivity due to the interface when that is not necessarily the case. The second option is calculationally burdensome, especially when there are multiple options for different storage configurations and hence many possible interfaces.

When an interface calculation is performed, essentially two semi-infinite arrays of each storage configuration are placed in the same model, possibly with a small gap between them in the case of rack-to-rack interfaces (i.e., no leakage is credited). If the model is sufficiently large enough (4 or more rows of storage cells of each configuration), the resulting k_{eff} of the interface cannot be less than the k_{eff} of the two individual storage configurations. Therefore, if the interface calculations show that the reactivity of the interface is essentially equivalent to the reactivity of the most reactive of the two storage configurations, then there is no additional neutronic coupling between the individual storage configurations. The following criteria are specified for interface calculations between two different storage configurations to be acceptable with no further restrictions:

- The calculated k_{eff} of the interface model is less than the maximum calculated k_{eff} of each individual storage configuration.
- If the interface calculation has a higher calculated k_{eff}, then restrictions should be specified to prevent these storage patterns from being used adjacent to one another.

In practice, interfaces show a higher reactivity than the individual storage configurations when high reactivity fuel is placed adjacent to one another across the interface. Care should be taken with interfaces to ensure that high reactivity fuel adjacent to one another across the interface is explicitly modeled and determined to be acceptable or not (if not, then restrictions should be specified to prevent these interfaces from occurring).

An interface can also occur between old and new racks. If the separation distance between the new and old racks is more than 6 inches at the interface, then there is no need to evaluate the interface between storage racks/configurations.

6.3 Abnormal and Accident Conditions

The licensee should consider all credible abnormal and accident conditions. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these abnormal and accident conditions, as long as the conditions do not also result in a dilution of soluble boron. The separate boron dilution accident is discussed in Section 7.3.

The following scenarios should be considered as part of postulated abnormal and accident conditions. Note that if a single accident scenario is clearly limiting, then other less limiting scenarios need not be explicitly calculated, but should be justified as being bounded. If the licensee determines that based on site specific rationale an accident condition is not credible, the submittal should include justification. If a design basis accident affects the inputs to the criticality analysis (e.g. if an earthquake results in physical changes to the neutron absorber material), then they should be considered.

6.3.1 Temperatures Beyond Normal Operating Range

The spent fuel pool has a normal operating range for the bulk temperature of the spent fuel pool water. Under accident conditions (loss of cooling) this temperature could be elevated beyond the normal operating range. Because the pool temperature is not a major contributor to reactivity and soluble boron credit can be taken for accident conditions, analysis should be done at boiling conditions with a void fraction up to 20% to confirm that higher temperature conditions are not limiting for PWR pools.

6.3.2 Dropped and Mislocated Assembly

A dropped fresh fuel assembly on top of the spent fuel rack can either land horizontally on top of the rack or vertically outside the rack. The horizontal drop is not the most limiting accident condition due to the separation between the dropped assembly and the active fuel provided by the structure above the active fuel. This separation prevents neutronic coupling but even if there is some coupling the other accident conditions are more limiting. Therefore, no analysis of a horizontal fuel assembly on the top of the rack is necessary.

A mislocated fresh fuel assembly outside and adjacent to the storage racks (inside the pool wall) should also be evaluated if applicable, unless there is not enough room to physically fit a fuel assembly in between the racks and/or the pool wall.

6.3.3 Assembly Misload

Misloading of a single fresh fuel assembly into an unapproved location should be evaluated as a postulated accident scenario. This accident scenario is postulated as an error on the part of the fuel crane operator to properly locate a fuel assembly in the correct storage location during fuel movement. For all storage configurations, an evaluation of a fresh fuel assembly of the maximum allowable enrichment, with no burnable absorbers should be evaluated in the storage location that provides the largest positive reactivity increase.

For BWRs spent fuel pools that contain a single region of uniformly loaded fuel with a given peak reactivity limit, the misload event does not need to be considered. If a BWR spent fuel pool has multiple regions with different peak reactivity limits, then a misloaded bundle with highest peak reactivity limit should be evaluated in the lower peak reactivity regions.

Additionally, there is the possibility of an error occurring in the selection of appropriate storage configurations such that a single initiation event can result in multiple fuel assemblies being misloaded. Whereas a single misload is typically a result of an error in the fuel handling selection or relocation of an assembly (i.e., picking up and moving an assembly other than the intended assembly), a single event resulting in multiple misloaded assemblies is typically the result of a planning or process error. Therefore, whether multiple misloaded assemblies is credible from a single event depends upon the administrative controls and processes the licensee establishes for assuring compliance with the loading patterns. Implementing a robust administrative control program for verifying used fuel assembly configurations and addressing potential non-compliant loading conditions therefore becomes vital to precluding a common cause failure of mis-loading multiple assemblies.

It is important to have a multi-tier defense-in-depth program in place to prevent or mitigate the severity of a scenario where multiple assemblies are located into the wrong storage locations. Specific aspects of this defense-in-depth program include the following:

- Licensee controlled procedures, programs
- Event tree analysis
- Post-movement fuel assembly verification
- Storage cell blocking devices
- Analysis of multiple misload scenarios, if applicable

Additional details of each of these elements are provided in the following sub-sections

6.3.3.1 Licensee Controlled Administrative Programs

The spent fuel pool criticality analysis specifies the acceptable storage configurations and limits on the type and characteristics of fuel (i.e., burnup, enrichment, cooling time, etc.) to ensure compliance with the acceptance criteria. Adherence to these requirements is accomplished by the licensee prior to any fuel movement to ensure that the fuel assembly is placed in an acceptable location. There are many commercial software packages available that can assist the licensee in determining the acceptability of a fuel assembly to be placed in a location in accordance with the Technical Specification and the spent fuel pool criticality analysis.

The use of a validated software package provides an additional barrier to prevent a common-fault error of selecting the wrong location for multiple fuel assemblies. Additionally, the following features should be implemented to reduce the risk associated with the incorrect placement of multiple fuel assemblies in the spent fuel storage racks:

- Production of reports that show acceptability of fuel assembly locations
- Graphical representation (fuel assembly burnup, enrichment, cooling time against the limits for the storage configuration) to augment manual verification
- Visual, color-coded spent fuel pool maps showing acceptability of fuel assembly locations
- Pre-verification of planned fuel moves
- Detailed administrative procedures for implementation
- Training and qualification of engineers responsible for spent fuel assembly selection and verification
- Independent verification of the validated software output, such as Fuel Transfer Logs (FTLs)
- Training of responsible engineers prior to implementation of new storage configurations or Technical Specification loading curves

6.3.3.2 Event Tree Analysis

An event tree graphically represents the various accident scenarios that can occur as a result of an initiating event (i.e., a challenge to plant operation). Toward that end, an event tree starts with an initiating event and develops scenarios, or sequences, based on whether a plant system succeeds or fails in performing its function. The event tree then considers all of the related systems that could respond to an initiating event, until the sequence ends in either a safe recovery or an accident event.

While an event tree analysis has not been historically applied to the credibility of an inadvertent criticality event in the spent fuel pool, there are several studies that have looked at the probability of a misloaded fuel assembly in a transport or storage cask [37, 40]. These studies can be used as guidance for creating an event tree analysis specific to a particular spent fuel pool configuration.

6.3.3.3 Post-Movement Assembly Verification

Verification of proper placement of fuel assemblies into approved storage locations after fuel movement can provide an independent confirmation of the acceptable storage configurations in the spent fuel pool. There are several potential processes that are suggested here that allow for additional defense-in-depth barriers to be implemented for ensuring proper placement of fuel assemblies:

- Visual verification of fresh versus spent fuel by fuel handling operators during fuel movement
- Administrative verification of high reactivity fuel assemblies prior to and after fuel movement.
- Post movement verification of fuel assembly locations

6.3.3.4 Storage Cell Blocking Devices

One simple approach to allow higher reactivity fuel to be placed in high-density racks is to designate specific storage cells to remain empty. However, placing either a fresh or spent fuel assembly in these storage locations under a multiple misload scenario would provide a significant reactivity addition. To prevent the misloading of multiple fuel assemblies into storage locations intended to be empty, blocking devices can be employed. Blocking devices are physical hardware installed into storage cells for the purposes of preventing the inadvertent placement of a fuel assembly into these locations. To ensure that maximum benefit and flexibility of these devices, the following criteria are recommended for blocking devices:

- Physically configured to prevent insertion of a fuel assembly in a fuel storage location,
- Requires specialized tools to install or remove the blocking device from a storage location,
- Designed to preclude falling inside a storage location or being dislodged from its position during normal operation,

- Contain a lock-in-place feature to prevent inadvertent movement,
- Support a load which will cause the underload trip sensor to activate. This is typically the load of one fuel assembly plus the handling tool,
- Allow for continued water flow through the storage cell.

Fuel-debris trash cans may be used as blocking devices, provided that they meet all of the above criteria except the requirement for specialized tools. Specialized tools are not required for trash cans as their physical appearance is easily distinguishable from a fuel assembly

Blocking devices do not need to be designed to prevent a dropped fuel assembly from entering the storage cell. However, the accident analysis must consider a single dropped fuel assembly in the storage cell with the blocking device. Multiple Misload Analysis

Some licensees may be able to demonstrate that a multiple misload from a single event is not credible, while others may determine it is credible and choose to analyze the consequences of a multiple misload. Again, the administrative controls and processes the licensee establishes for assuring compliance with the loading patterns will influence the potential consequences of a multiple misload from a single event. For example, a process check to ensure that a fresh fuel assembly is not selected when a used fuel assembly is intended to be selected (perhaps by confirming the physical appearance of the assembly) could eliminate the need to assume a multiple misload of fresh fuel. In this example, the misloaded fuel assemblies could represent the minimum burnup for once burned fuel with the highest enrichment, since theprocedural check would ensure that it is not credible to misload fresh fuel assemblies.

6.3.4 Seismic Movement

The spent fuel racks are designed to withstand the ground motions associated with the design basis seismic event. However, the spent fuel racks may sway or slide slightly in the spent fuel pool. These motions are small and do not have a significant effect on reactivity. Typically, the spent fuel rack baseplate is designed to prevent the racks from coming too close together or the walls from being damaged during seismic events. A straightforward approach for addressing seismic shifting is to assume that the racks are moved as close together as possible as allowed by the baseplate. For most BWR spent fuel pools, the analysis is performed assuming as infinite array, so seismic issues do not require additional analysis or justification.

7 SOLUBLE BORON CREDIT

7.1 NORMAL CONDITIONS

10CFR50.68 [1] allows soluble boron credit of up to 5% Δk. That is, if credit is taken for soluble boron, k_{eff} of the spent fuel pool must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. Analyses performed in accordance with the guidance in Sections 5.1 and 5.2, including unborated water, must ensure that the maximum calculated k_{eff}, including all biases and uncertainties meet the k_{eff} limit of less than 1.0. The criticality safety analysis must also demonstrate that if the spent fuel pool is flooded with borated water, k_{eff} must not exceed 0.95, at a 95% probability, 95% confidence level.

7.2 Accident Conditions

For conditions with soluble boron, the accident conditions in Section 6.3 should be evaluated at the minimum soluble boron concentration allowable under the site's Technical Specification. In other words the accident condition does not need to consider a simultaneous boron dilution event, per the double-contingency principle, if the accident does not also result in boron dilution. This is justified through application of risk insights, in that the probability of a significant boron dilution event (violating the minimum pool soluble boron concentration) is remote, and that there have not been any known cases of its occurrence in the history of nuclear power operations.

For the accident conditions, the maximum calculated k_{eff}, including all biases and uncertainties, must be less than the regulatory k_{eff} limit of 0.95.

7.3 Boron Dilution

In the event the licensee is crediting soluble boron in the criticality safety calculation, a boron dilution accident should be considered. The boron dilution analysis should initiate at the minimum allowable normal soluble boron concentration as described in the plant Technical Specifications and is consistent with the boron concentration assumed in the criticality analysis to maintain subcritical conditions (0.95) for normal conditions. The boron dilution analysis should confirm the time needed for dilution to reduce the soluble boron concentration (from the plant technical specification concentration to the boron concentration assumed in the criticality analysis which shows that for normal operation the k_{eff} is less than 0.95) is greater than the time it needed for actions to be taken to prevent further dilution.

A graded approach to the boron dilution analysis may be taken depending on the amount of soluble boron being credited versus the amount required to be in the spent fuel pool. For example, if 2000ppm of soluble boron is required by the Technical Specification, and a licensee that takes credit for nearly this entire amount (e.g. 1800ppm) in the criticality analysis, then the licensee would need to provide additional justification for their assumptions over a licensee that only takes credit for a small portion of soluble boron (e.g. 200ppm). Similarly, an analysis that determines that the credible soluble boron dilution event would not reduce boron below the required amount in less than 24 hours, would not need to provide additional justification for the assumptions in the boron dilution analysis if it can be credibly shown that action would be taken to prevent further dilution in less than 8 hours.

8 CALCULATION OF MAXIMUM K_{EFF}

The maximum k_{eff} must be determined for the spent fuel pools and new fuel vaults including uncertainties and biases. The maximum k_{eff} is determined by adding to the nominal calculated k_{eff} any biases that may exist in the methodology and the applicable uncertainties using the formula described in Equation 1:

$$k_{max} = k_{eff} + \sum_{i=0}^{m} Bias_i + \sqrt{\sum_{j=0}^{n} Uncertainty_j^2} \quad \text{(Equation 2)}$$

As can be seen from the above expression, uncertainties are statistically combined (assuming that such uncertainties are mutually independent) while biases are summed up. The biases and uncertainties that should be included are discussed within applicable sections of this document. These are summarized here for completeness:

Biases
Criticality Code Validation bias
Moderator Temperature
Fuel assembly design

Uncertainties
Fuel Manufacturing tolerances
Rack Manufacturing tolerances
Depletion Code Uncertainty
Burnup Uncertainty
Criticality Code Validation Uncertainty

Uncertainties should be determined for the proposed storage facilities and fuel assemblies to account for tolerances in the mechanical and material specifications. An acceptable method for determining the maximum reactivity may be either (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff}, or (2) a sensitivity study of the reactivity effects of variations of parameters within the tolerance limits. If used, a sensitivity study should include all possible significant allowable variations within the material and mechanical specifications of the fuel and racks; the results may be combined statistically provided they are independent variations. Combinations of the two methods may also be used.

9 LICENSEE CONTROLS

9.1 LICENSEE CONTROLS

A licensee should establish controls that help to ensure that the conditions evaluated in the nuclear criticality safety analysis are and remain bounding to the current plant operating parameters. Appropriate licensee controls include plant procedures and programs that control storage configurations, and burnup/enrichment loading curves, and ensure that the storage of fuel is bounded by the criticality analyses.

9.2 PROCEDURAL CONTROLS

A licensee establishes procedural controls in order to ensure that used fuel is stored in accordance with the Technical Specifications, and to govern the planning and performance of fuel movements. These procedures implement the requirements for tracking the location of fuel assemblies in accordance with Special Nuclear Material (SNM) regulations and the spent fuel pool criticality analysis. They also ensure proper assembly selection for core loading activities, thermal management, gamma flux, etc. In addition programs and procedures are established to ensure that the licensee is following their software QA plan. The software QA program covers the use of codes for criticality analyses and software used to plan and implement fuel movements.

Procedural controls should be developed based upon the complexity of storage patterns in order to provide reasonable assurance of adequate public health and safety. The procedures may also affect the assumed accident conditions (see Section 6.3) For example, if storage patterns are relatively straight-forward and the procedures preclude a credible multiple misload event resulting from a single initiating event, then the multiple misload event would not need to be evaluated as an accident condition. The following are typical procedures used by licensees. Additional procedures should be considered for more complex storage patterns.

- Pool Assembly Storage Planning
  - Fuel Characterization
    - Fuel reactivity category determination, e.g.,
      - Burnup
        (e.g., plots of burnup v enrichment to identify outliers, possible errors)
      - Enrichment
        (e.g., plots of burnup v enrichment to identify outliers, possible errors)
      - Decay time
    - Component inserts
  - Development of planned pool fuel assembly storage configurations
    - Use of verified software application to confirm planned pool configuration is in accordance with the criticality analysis
    - Independent verification of desired pool configuration
  - Development of Fuel Transfer Forms (FTF) to implement planned storage configuration
    - Use of verified software application to generate FTFs
    - Independent verification of FTFs

- Fuel Movement
  - Use of only approved FTFs
  - Activities of the Fuel Mover
  - Independent verification
    (the verifier should have no concurrent duties)
  - Independent FTF Step Verifier
    (the step verifier should have no concurrent duties)
  - Continuous communications between fuel mover, verifier, and step verifier
  - Personnel Training
  - Pre-job briefs

- Spent Fuel Pool
  - Bounding soluble boron requirement
    (use of a larger soluble boron concentration to provide more reactivity hold-down to minimize the effect of assembly misloadings)
  - Technical Specification for soluble boron surveillance
  - Neutron Absorber Panel material behavior monitoring program
- Software Requirements:
  - Independent review of software implementation and revision, testing and documentation is performed by an independent reviewer
  - Configuration controls to ensure integrity of executable files and data files
  - Cyber security controls prevent tampering / inadvertent changes

- Database Requirements:
  - Independent review and approval of all database updates
  - Procedures to ensure integrity of database prior to utilizing the data

9.3 New (Future) Fuel Types

It is common for licensees to periodically use newer fuel types that have more desirable in-reactor performance characteristics. However, it is impossible to predict the characteristics of fuel types that may be used in the distant future at the time of developing an application involving fuel criticality analyses. Therefore, the licensee should implement a process to assess (or check) newer fuel designs as they are implemented to ensure they are bounded by the existing design basis/analysis of record for the spent fuel storage rack or new fuel vault.

If an initial assessment determines that the new fuel type represents a potential change to the existing criticality safety design basis/analysis of record for the storage rack or vault, then a full criticality analysis should be performed. In accordance with 10 CFR 50.59, the full criticality analysis of the new fuel should include all credible configurations that have previously been analyzed for existing fuel types (e.g. normal, off-normal, and accident conditions) and interfaces with other fuel types.

The 10 CFR 50.59 [8] process is used to determine whether NRC review and approval is necessary prior to implementing the new fuel design.

9.4 Pre- and Post-Irradiation Fuel Characterization

Fuel characterization is the process of ensuring that the actual nuclear fuel assemblies to be stored are bounded by the criticality analysis assumptions. This process should involve comparing actual fuel assembly and operating parameters to key assumptions utilized in the criticality analysis, and require further evaluation if the assumptions are not met. The intent is to ensure that changes in fuel design, core design, or cycle operation (both anticipated and unanticipated) are properly evaluated prior to storing the fuel.

Note that fuel characterization as discussed in this section is separate from the typical categorization of fuel assemblies according to initial enrichment, assembly-average burnup, and, in some cases, cooling time, that is used to determine where fuel assemblies may be placed in the spent fuel pool.

For any given fuel assembly, fuel characterization consists of two processes. The first process is pre-irradiation characterization, and its purpose is to review the design of the fuel assembly against the parameters assumed in the criticality analysis. Ideally, this is performed as part of the core design process. In any case, it is performed before the fuel in question is placed, for the first time, in the new or spent fuel racks. For pressurized water reactors, the key inputs pertain to the fuel loading (fuel pellet mass, diameter, density, enrichment, etc.) and to the fuel-to-moderator ratio (fuel rod diameter, fuel rod pitch, etc.). Boiling water reactors should also consider the lattice itself (8x8, 9x9, 10x10, etc.), as well as the characteristics of the fuel channel. One acceptable method for BWR fuel characterization is the in-core $k\infty$ methodology. This method establishes infinite-lattice reactivity limits for each fuel storage region as part of the criticality safety analysis. Each unique fuel design is then validated against this reactivity limit to establish its acceptability for storage. Other characteristics to be considered will depend upon the nature of the criticality analysis itself. For example, if the analysis took credit for the initial presence of burnable absorbers in the fuel, then the characteristics of the burnable absorber (type, loading, and configuration) should also be considered.

The second process, called post-irradiation characterization, is only applicable if the criticality analysis in some way credits the in-reactor depletion of the fuel assemblies (i.e., burnup credit). If burnup is credited, a process should be implemented to ensure that the fuel was depleted in a manner consistent with the assumptions in the criticality analysis.

Post-irradiation characterization is concerned with ensuring that certain parameters assumed in the criticality analysis do, in fact, bound the actual operating history of the fuel assemblies. Parameters to be considered will depend on the methods and assumptions of the analysis. Some licensees may be able to verify that the reactor operated within the bounds of the analysis based on comparison to past operation, while others may need to verify more detailed reactor parameters or assembly specific parameters, such as:

- Axial burnup shape (if using Option 3 in Section 5.1.3)
- Moderator temperature (burnup averaged)
- Fuel temperature (burnup averaged)
- Soluble boron (burnup averaged)
- Control rod insertion
- Burnable absorber presence (particularly if discrete, removable burnable absorbers are used)

Ideally, the process of post-irradiation characterization is initiated as part of the core reload design process, so that potential non-compliances with the criticality analysis can be identified early on, and possible changes to the fuel or core design can be made to mitigate the concerns. Post-irradiation characterization should be finalized following actual reactor operation, to ensure that there were no significant operating differences from that assumed during the core reload design process. In particular, a re-evaluation of the post-irradiation characterization should be considered if such differences result in a significant hardening of the neutron spectrum experienced by fuel assemblies or alter the axial power shape in the fuel assemblies long enough to significantly impact the axial burnup shape of the fuel at discharge. Specifically, this could include:

- Operation for a significant period of time at reduced power or with control rods inserted more than during normal operations
- Changes to plant configuration that result in higher-than-expected reactor coolant temperatures

For both pre- and post-irradiation characterization, any differences that are not explicitly bounded by the criticality analysis should be evaluated to determine if there is any impact on the criticality analysis, in accordance with other licensee programs (e.g., 10 CFR 50.59 [8] process, operability evaluation).

9.5 NEUTRON ABSORBER MONITORING PROGRAMS[1]

Neutron absorbers serve as an important material to control reactivity in most spent fuel pool storage racks. As neutron absorbers significantly reduce reactivity, it is important to ensure that they continue to provide their criticality control function for the duration that they are relied upon in the criticality analyses. Neutron absorber monitoring programs should be developed with the purpose of ensuring that the neutron absorbers continue to provide the criticality control relied upon in criticality analyses. To accomplish this, the monitoring program must be capable of identifying whether unanticipated changes are occurring, and if anticipated changes are occurring that the anticipated characteristics of change can be verified.

Coupon testing, in-situ measurement or a combination of both are acceptable approaches to a neutron absorber monitoring program; however, alternative approaches are also acceptable if adequately justified. A monitoring program should also consist of identifying material testing, R&D and operating experience at other plants, and evaluation on the relevance of outside data on the in-service material. Acceptance criteria should be developed as the basis for the comparison of results in order to determine whether material performance is acceptable or actions are necessary to address performance issues.

9.5.1 Coupon Testing Program

Use of coupons is the preferred method for a neutron absorber monitoring program. Coupon testing programs should meet the following criteria:

- The number of coupons should be sufficient to provide sampling at an appropriate interval for the intended life of the neutron absorber. The intended life of the neutron absorber should be based upon the amount of time the neutron absorber will be relied upon to provide criticality control. This is typically the life of the plant (including license renewal) plus some additional time to permit off-loading the spent fuel pool during decommissioning.

- Sampling intervals should be based upon the expected rate of material changes, which may be influenced by the qualification testing of the material. For new materials that do not have a lot of operating experience in conditions similar to the pool environment (i.e. their ability to perform is not well known), the initial interval should not exceed 5 years, with subsequent intervals up to 10 years. For materials that have been used for several years in conditions similar to the pool environment (i.e. their ability to perform is well known), and for which stability in the material condition has been documented, initial and subsequent intervals up to 10 years is acceptable.

- Coupon testing can be categorized as basic or full testing. The coupon testing is used to identify whether unanticipated changes are occurring, and if they are, the condition of the

[1] While these guidelines for neutron absorber monitoring programs are intended for initial license applications and license amendment requests that install new neutron absorber materials, they may be useful for licensee's consideration in license renewal applications under 10 CFR Part 54.

neutron absorber material. The extent to which each of these is utilized should be determined based upon the operating history of the material, as follows:

a) Basic testing consists of visual observations, dimensional measurements, and weight. These parameters focus on identification of whether changes are occurring in the materials. Basic testing is appropriate when previous testing and operating experience of the material indicates that there are no degradation mechanisms that would result in loss of ^{10}B areal density that would affect reactivity.

b) Full testing may consist of a combination of density measurements, ^{10}B areal density measurements, microscopic analysis, and characterization of changes, in addition to the basic testing parameters. These parameters focus on quantifying changes if they are occurring in the materials. Full testing should be performed for the first coupon test, but may not be necessary for subsequent test periods unless a loss of ^{10}B areal density is anticipated. Basic testing may be used in combination with full testing for materials that have degradation resulting in loss of ^{10}B areal density to extend the interval of full testing, if appropriately justified.

- Coupons should be located such that their exposure to parameters controlling change mechanisms (e.g., gamma fluence, temperature) is conservative or similar to the in-service neutron absorbers.
- Results are acceptable to confirm the continued performance of neutron absorber materials if either:

  a) For materials that are not anticipated to have a loss of ^{10}B areal density; the ^{10}B areal density of the test coupon is the same as its original ^{10}B areal density.

  b) For materials that are anticipated to have a loss of ^{10}B areal density; the loss of ^{10}B (difference between ^{10}B areal density of the test coupon and its original ^{10}B areal density) is less than the loss of ^{10}B areal density used in the criticality analysis.

9.5.2 In-situ Measurement Program

In-situ measurement is another acceptable method for confirming ^{10}B areal density of neutron absorber material. In-situ measurement is used to identify whether unanticipated changes are occurring, and if they are, the condition of the neutron absorber material. There are two potential uses for in-situ measurements:

1. Supplement coupon monitoring to extend the coupon testing interval or permit greater reliance on basic testing.
2. In lieu of coupon testing if coupons do not exist.

Both uses of the in-situ measurement should meet the following criteria:

- In-situ measurement campaigns should be performed on an adequate number of panels and at an acceptable interval.
- Number of panels tested should be an appropriate statistical sample.

- Sampling interval is based upon the expected rate of material change, which may be influenced based upon the qualification testing of the material. For new materials that do not have a lot of operating experience in conditions similar to the pool environment (i.e. their ability to perform is not well known), the initial interval should not exceed 5 years, with subsequent intervals up to 10 years. For materials that have been used for several years in conditions similar to the pool environment (i.e. their ability to perform is well known), and for which stability in the material condition has been documented, initial and subsequent intervals up to 10 years is acceptable.

- Note that sampling interval can be longer if used in conjunction with coupons.

- Sources of measurement uncertainty should be identified and the degree of uncertainty quantified.

Additional criteria for in-situ measurements depend upon the performance of the neutron absorber material, specifically whether material changes result in a degradation of the ^{10}B areal density.

A. For materials where potential change mechanisms do not result in a loss of ^{10}B areal density, in-situ measurements are used to confirm their presence and provide validation of the original as-manufactured areal density. Results confirm the continued performance of neutron absorber materials if the <u>nominal</u> measured ^{10}B areal density is greater than the ^{10}B areal density assumed in the criticality analysis.

B. For materials where degradation mechanisms may result in a loss of ^{10}B areal density, in-situ measurements are used to determine the amount of ^{10}B areal density remaining. Results confirm that potential loss of ^{10}B has not resulted in the loss of the neutron absorber material's ability to perform its criticality control function if the nominal measured ^{10}B areal density minus the measurement uncertainty is greater than the ^{10}B areal density assumed in the criticality analysis.

9.5.3 Evaluating Neutron Absorber Test Results

Results from neutron absorber monitoring fall within the broad categories of 1) confirmation that no material changes are occurring, 2) confirmation that anticipated changes are occurring, and 3) identification that unanticipated changes are occurring. Processes should be established to evaluate results of the monitoring program with the criticality analysis input. If no changes, or if anticipated changes are occurring, then the material condition continues to be adequately represented in the criticality analysis.

If unanticipated changes are identified (either new mechanisms or anticipated mechanisms at rates or levels beyond those anticipated), then additional actions may be necessary. In addition to relevant regulatory and licensing processes (e.g. operability determination, reporting requirements, the 10 CFR 50.59 [8] process), the following technical assessments may be necessary.

- Determine if unanticipated changes could result in a loss of ^{10}B areal density. This is considered the only major impact to criticality control (See Section 5.2), since ^{10}B areal density has a much larger impact than dimensions. Evaluation of the effects of ^{10}B areal density on the criticality analysis should be performed and addressed through appropriate licensee processes.

- Determine if unanticipated changes not resulting in loss of ^{10}B areal density have an impact on the criticality analyses. Dimensional or non-neutron absorbing material changes (e.g. formation of gaps, localized displacement of moderator, or superficial scratches) may have no or little impact on the criticality analyses. However, the potential effects of these changes on the criticality analysis should, nevertheless, be evaluated and addressed through appropriate licensee processes.

10 REFERENCES

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2. Title 10 of the *Code of Federal Regulations* (10 CFR) 70.24, Criticality Accident Requirements.

3. Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix A, General Design Criteria for Nuclear Power Plants Criterion 62, Prevention of Criticality in Fuel Storage and Handling.

4. Title 10 of the *Code of Federal Regulations* (10 CFR) 52, Licenses, Certifications, and Approvals for Nuclear Power Plants.

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APPENDIX A: COMPUTER CODE VALIDATION

A.1 FRESH FUEL CRITICALITY CODE VALIDATION

The criticality computer codes used for the criticality safety analysis should be validated using measured data. This validation should consist of five steps:

1. Identify range of parameters to be validated
2. Select critical experiment data
3. Model the experiments
4. Analyze the data
5. Define the area of applicability of the validation and limitations

NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," provides guidance on one approach for performing the validation [13].

A.1.1 Identify Range of Parameters

The first step is to identify the range of parameters to be validated. Examples of parameters that should be selected include type of fissile isotope, enrichment of the fissile isotope, fuel chemical form, etc. These selected parameters will lay the foundation for determining the area of applicability of the validation. Specifically the neutronic behavior is influenced by the following parameters, which should be covered by the selected experiments:

- Isotopic Content
    - Experiments should cover material for the rack structure (e.g., stainless steel), material for the cladding (e.g., zirconium), fissile isotopes in the applicable enrichment range (e.g., U-235 for low enriched UO_2, Pu-239 for MOX), water, and others if applicable: boron for the soluble boron and absorber plates, gadolinium if peak reactivity is used (BWRs) or if credit for gadolinium in fresh fuel is used, and/or silver/indium/cadmium if control rods are used in the criticality analysis.
- Spectrum
    - The spectrum can be affected by geometry and storage rack design (e.g., a region with a flux trap design or a region with no flux traps), therefore, the critical experiments should cover a range of spectra. The spectrum range can be quantified by an index such as the energy of the average lethargy of neutrons causing fission (EALF) or average energy group causing fission (AEG). Historical indices used include hydrogen-to-fissile atoms ratio (H/X), and fuel-to-moderator ratio.

- Geometry
  - Key geometric features are the fuel pin pitch, pellet or clad diameter, assembly separation, and boron areal density.

A.1.2 Selection of Critical Experiments

The features listed above are covered with available critical experiments, for example the OECD/NEA *International Handbook of Evaluated Criticality Safety Benchmarks Experiments* [26] and the HTC critical experiments [14] are considered an appropriate reference for criticality safety benchmarks. The handbook has reviewed the benchmarks and carefully evaluated the uncertainties in the experiments. Other sources for critical experiments may also be acceptable and should include an estimate of the uncertainty in the critical experiments.

The set of experiments selected should ensure a statistically appropriate validation. Care should be taken in selecting critical experiments so that trends can be identified and addressed.

A.1.3 Modeling the Experiments

Section 2.3 of NUREG/CR-6698 [13] states that it is acceptable to "choose to use input files generated elsewhere to expedite the validation process". It should be emphasized, however, that although the input files may initially come from somewhere else, the modeling of the critical experiments should match, as closely as possible, the modeling used in the criticality safety analysis (e.g. comparable level of geometric modeling detail).

A.1.4 Analysis of the Critical Experiment Data

NUREG/CR-6698 [13] defines the steps of "Analyze the data" as:

1. Determine the Bias and Bias Uncertainty
2. Identify Trends in Data, Including Discussion of Methods for Establishing Bias Trends
3. Test for Normal or Other Distribution
4. Select Statistical Method for Treatment of Data
5. Identify and Support Subcritical Margin
6. Calculate the Upper Safety Limit

NUREG/CR-6698 [13] provides equations for the determination of the bias and bias uncertainty. These equations weight the experiments by the experimental uncertainty. It is important that the experimental uncertainty is reasonable to ensure meaningful trend analysis. It is noted that inaccurate experimental uncertainties may result in inaccurate trend results. The uncertainties provided in the OECD criticality benchmark handbook [26] are sufficient for this purpose so the statistical approach defined in NUREG/CR-6698 [13] should be used.

It is important to look over the calculated biases for trends in the data. At a minimum statistical analysis should be performed to check for a trend in the bias due to differences in spectrum and enrichment. Seeking more trends is recommended. However, it is noted that trends in some parameters may actually be the result of trends in spectrum or enrichment, i.e. dependent parameters that are embedded in the data. In these cases, only spectral or enrichment trends need to be considered.

The equation in Section 2.2 can be used to calculate the maximum k_{eff}. Alternatively, the method in NUREG/CR-6698 [13] for determining an upper safety limit on k_{eff} which includes the uncertainty determined from the critical experiments may be used. The uncertainties from the critical benchmark analysis can be statistically combined with other uncertainties such as manufacturing tolerances (see Section 2.2). The bias and uncertainty determined from the critical experiments may be applied as a function of the trending parameters or as conservative values that cover the desired range(s).

A.1.5 Area of Applicability

The validation of the calculational methodology for nuclear criticality safety analyses covers an area of applicability, or also known as the "benchmark applicability". [10] The criticality safety analysis should define and document this area of applicability.

The following subsection provides further detail and guidance of how to apply and use the area of applicability in the nuclear criticality analysis.

Limitations and Conditions

In the validation, a range of parameters should be established that are important to criticality and that reflects the range of conditions, normal and abnormal, that the fuel assemblies could experience in the new fuel vault and the spent fuel pool. Parameters, per ANSI/ANS-8.24, that should be considered include [10]:

- Nuclide composition and chemical form of all associated materials;
- Geometry (e.g., lattice pattern, spacing, reflector location, size, shape, and homogeneity or heterogeneity of the system); and
- Characterization of the neutron energy spectrum.

Again, the selection of the range of these parameters should be determined based on both normal and credible abnormal new fuel vault and spent fuel rack conditions.

Trend Evaluation

Part of the validation is to identify whether the bias has a dependency on any of the parameters in the area of applicability. The parameters selected for trending evaluation should be based on the characteristics of the system under consideration. [10]

If a significant trend exists in a bias of an important parameter in the validation of the code, then the criticality safety analyses should appropriately address the trend when determining the appropriate bias and uncertainty to utilize.

Extrapolation

If the experiments do not fully cover the analyzed system, then it may be possible to extrapolate the validation. The area of applicability may be extended beyond the range of experimental conditions by employing the trends in the bias. NUREG/CR-6698 [13] provides further guidance for extending trends and accounting for increasing uncertainty if there are insufficient critical experiments.

A.2 Used Fuel Depletion code Validation

Additional validation is required for used fuel since it depends on depletion analysis and the reactivity worth of isotopes not found in the fresh fuel critical experiments. This section provides several validation approaches for both PWR and BWR racks to explicitly quantify a depletion uncertainty. The use of the 5% Δk as a depletion uncertainty is acceptable for ENDF/B-V through VII. [27, 33, 34].

A.2.1 PWR Used Fuel Validation

Two acceptable approaches for PWR used fuel validation are presented. These approaches can be combined or used separately. The approaches are:

1. Use benchmarks based on depletion reactivity inferred from reaction rate measurements at power plants.
2. Use chemical assays and cross section uncertainties.

A.2.1.1 Validation Using Measured Flux Data from Power Reactors

PWR depletion benchmarks were developed by EPRI [27,28] using a large set of power distribution measurements to ascertain reactivity biases. The predicted reactivity of the fuel assemblies was adjusted to find the best match between the predicted and measured power distribution. EPRI used 680 flux maps from 44 cycles of PWR operation at 4 PWRs to infer the depletion reactivity [28]. The depletion reactivity has been used to create 11 benchmark cases for various burnups up to 60 GWd/T and 3 cooling times 100 hour, 5 years, and 15 years. All of these benchmark cases should be analyzed with the code set (depletion and criticality codes) to be used in the criticality analysis to establish a bias for the depletion reactivity. The uncertainty in the benchmarks should be used as the depletion reactivity uncertainty. These biases and uncertainties cover both the isotopic content uncertainty and the worth uncertainty associated with depletion. They account for all the changes from the initial fresh fuel condition. The bias and uncertainty associated with fresh fuel are also required to be included in the validation of the criticality safety evaluation. A companion EPRI report describes in detail how to apply the benchmarks in the criticality safety analysis [27].

A.2.1.2 Validation Using Chemical Assays and Worth Experiments

Depletion validation using chemical assays and worth experiments, performed for the NRC by ORNL, are documented in NUREG/CR-7108 [16] and NUREG/CR-7109 [15]. NUREG/CR-7108 evaluates differences in measured and calculated isotope concentrations. NUREG/CR-7109 evaluates the bias and uncertainty attributable to cross section data uncertainty for minor actinides and fission products.

The NUREG/CRs include biases and uncertainties that can be used in the validation of PWR and BWR criticality analyses if the system and method are similar to those used to produce the bias and uncertainty. It should be noted that depletion worth bias and uncertainty derived from chemical assay data tend to be significantly conservative due to the large experimental uncertainties in performing chemical assays. Experimental uncertainty in measured isotopic content is carried over into the calculation of measured versus predicted SFP rack K_{eff}, thereby increasing the apparent uncertainty of predicted depletion worth. Studies in the past have shown that predicted depletion uncertainty derived from chemical assay data changes very little with calculational method changes, which would be expected if the uncertainty is dominated by the uncertainty in the chemical assays rather than the uncertainty from the calculational method.

In NUREG/CR-7108 a method was presented [16] that relies on determination of distribution functions (measured versus predicted isotopic content) for key isotopes. This Monte Carlo approach used large burnup bins in order to get enough data to establish the distribution of data around the mean for each isotope. Although this appropriately accounts for the variation in number of isotopes included in the chemical assay samples, it loses most of the burnup dependence of the data, resulting in very large relative uncertainty for burnups below 30 GWD/MTU.

A direct difference approach was also presented in NUREG/CR-7108 that directly models each chemical assay sample. Direct difference modeling does not lose the burnup dependence of the data and handles the missing isotopic data by using "surrogate data" for nuclides without measurements. If validation through chemical assays is selected, it is recommended that the 99 of the 100 chemical assays selected for NUREG/CR-7108 [16] be analyzed and then the direct difference approach be applied to determine a bias and uncertainty as a function of burnup. Note that the HB Robinson sample N-9C-J should not be included in the set since it was deliberately taken from under an Inconel grid.

Both of these chemical assay approaches result in a conservative estimate of the bias and uncertainty for the 28 major isotopes selected. However, NUREG/CR-7108 states that most of the bias and uncertainty (90-95%) is attributable to U-235 and Pu-239. It has been shown that the isotopes in excess of the 28 major isotopes selected in NUREG/CR-7108 have a relatively small worth. Therefore, for analyses crediting more than the 28 major isotopes, it is recommended that the bias and uncertainty from the chemical assays be applied for all isotopes.

NUREG/CR-7109 [15] recommends a bias of 1.5 % of the reactivity worth of the isotopes not included in benchmark critical experiments to cover cross section bias and uncertainty. The isotopes used in addition to the 28 isotopes directly evaluated are expected to behave similarly so a bias of 1.5% of the reactivity worth of all depletion isotopes except U, Pu, and Am-241 is recommended. This recommendation applies for calculations using ENDF/B-VII cross sections. Additional justification should be provided for evaluations using other cross section data.

This approach assumes that the major actinides have been validated using critical experiments. Therefore MOX critical experiments are needed. For ENDF/B-V and VII MOX critical experiments (including the HTC critical experiments) have as smaller bias than fresh UO_2 experiments. Based on this observation fresh the bias and uncertainty from fresh UO_2 experiments may be used and analysis of MOX criticals is not needed for ENDF/B-V or ENDF/B-VII.

A.2.2 BWR USED FUEL VALIDATION

An acceptable approach for BWR used fuel validation is presented.

A.2.2.1 Validation Using Measured Critical Data from Power Reactors

Each time a BWR is loaded with fresh fuel during an outage, a cold critical control rod configuration is predicted using a lattice physics and core simulator code package. To assess the accuracy of depletion codes in calculating used fuel isotopes and their corresponding reactivity, the criticality analyst can compare critical conditions from power plant startups with predicted eigenvalues. Control rods are then withdrawn from the core using the prescribed sequence until the core reaches a critical state. The core period, temperature, and control rod positions are then fed back into the lattice physics/core simulator package to obtain the calculated eigenvalue for the measured critical configuration.

The use of such measured critical data is applicable because the cold critical conditions are very similar to the rack conditions in that:

1. The moderator temperature and density is about the same as the rack,
2. The control rods which are being removed during the startup are similar (e.g. in their neutronic effects) to absorber plates in rack,
3. The fuel itself is the same (pellet diameter, pin diameter, rod pitch, etc), and
4. The average burnup is similar to the peak reactivity burnup used in the criticality analysis.

As the core is in a cold, unvoided, mostly controlled state for these measurements, the variability of the measured eigenvalue to factors other than isotopic variations in the fuel (such as fuel temperature, moderator temperature, power density, instantaneous void fraction, etc.) is minimized. Additionally, as the cold critical measurements either involve a small local subset of control rods and their adjacent bundles or typical control rod withdrawal sequences involve banked rod movements to significantly extracted positions at several distinct and spatially separate locations in the core, the results of the corresponding calculation will be sensitive to the fidelity of the lattice physics code in assessing local isotopic compositions and reactivities. Thus, measured critical conditions are capable of providing benchmark experiments for spent fuel pool conditions.

By comparing the measured data to calculated results over a large range of startup experience, a bias (Δk_{SUb}) and bias uncertainty (Δk_{SUu}) can be assessed for the lattice physics/core simulator package. The following are two approaches to use this bias and bias uncertainty.

A.3 Application of Code Validation

Method A (Assigning the startup bias and uncertainty to isotopic content only):

In Method A, the criticality code validation, isotopic composition, and cross section uncertainties are assessed in three steps:

1) The criticality analysis code and fresh fuel isotopic cross sections are validated using fresh fuel critical experiments as described in Section 3.2.1.. The inclusion of the HTC critical experiments in the fresh fuel validation can cover the major actinide worth uncertainty.

2) The measured startup core critical bias (Δk_{SUb}) and bias uncertainty (Δk_{SUu}) is applied to cover the isotopic composition uncertainty. This is appropriate since isotopic content for the criticality analysis comes directly from the same lattice physics code used for the reactor startup analysis, and the corresponding startup bias and bias uncertainty are a function of the lattice physics code's capability to calculate nodal cross sections (and isotopics) for the core simulator.
3) Actinides' and fission products' cross sections which are not explicitly represented in the critical experiments are covered by adding an uncertainty that is proportional to the reactivity worth of the isotopes not explicitly validated. One approach to do this has been developed in NUREG/CR-7109 [15].

The final validation bias in Method A is the sum of the bias from the startup data and the bias from the fresh fuel critical experiments. The uncertainty is the statistical combination of the uncertainties from the startup data, the fresh fuel critical experiments, and proportional reactivity worth assessment. These uncertainties can also be statistically combined with other independent uncertainties such as rack and fuel manufacturing tolerances.

Method B (Reactivity use of the startup bias and uncertainty):

In Method B, it is assumed that the measured core critical bias (Δk_{SUb}) and bias uncertainty (Δk_{SUu}) fully validates the lattice physics code results and therefore covers both the uncertainty in isotopic content and worth. To implement, a reactivity equivalence must be established between the lattice physics code used for the depletion analysis for the startups and the Monte Carlo code used in the criticality analysis.

In Method B the following steps are required:

1. Using the same lattice physics code as used in the core startup analyses a calculation of the k-inf at the peak reactivity condition (enrichment, burnup, and gadolinium) is performed.
2. This same peak reactivity condition is modeled in the criticality Monte Carlo code to establish a bias (Δk_{MC}) between the Monte Carlo code and the lattice physics code. Notice that the isotopic content comes from the lattice physic code depletion.
3. Since the power reactor startups do not have stainless steel and possibly other rack features, fresh fuel critical experiments have to be run to seek any bias and uncertainty from these features. Fresh fuel critical experiments also validate the criticality analysis tool's solution method, as described in Section 3.2.1 of this guidance. (Note that this step in practice is the same as step 1 of Method A but with slightly different justification.) This analysis results in a bias (Δk_{cb}) and uncertainty (Δk_{cu}). Since the power reactor startup bias and uncertainty contain uranium and plutonium the bias due to these isotopes are counted twice. To assure no cancelation of errors negative, biases are ignored. Since the power reactor startups contain fuel with well understood fission product content, no additional bias and uncertainty for fission products is needed.

The final validation bias in Method B is the sum of the bias from the power reactor startup data (Δk_{SUb}), the bias from the benchmarking of the Monte Carlo code to the lattice physics code (Δk_{MC}), and the bias from the fresh fuel critical experiments (Δk_{cb}). The uncertainty is the statistical combination of the uncertainties from the startup data (Δk_{SUu}) and the fresh fuel critical experiments (Δk_{cu}). These uncertainties can also be statistically combined with other independent uncertainties such as rack and fuel manufacturing tolerances.

A.4 Alternate Code Validation

If a code (the primary code) is not capable of directly modeling the benchmark experiments, then an intermediary code (i.e., a secondary code) may be used that is validated to the benchmark experiments, and to which the primary code is validated. The primary code (code used for the criticality safety analyses) should still be capable of accurately modeling all the important neutronic and geometric aspects of storage. The secondary code should be validated by benchmarking to experiments that are similar to the neutronics and geometry of the criticality safety analysis. The primary code can then be validated by benchmarking to the secondary code over a range of parameters (neutronic and geometric) that bound the range of parameters for the criticality safety analysis. Those parameters that are important to be validated between the primary and secondary code include:

- Enrichment,
- Burnup,
- Absorber areal density,
- Soluble boron content, and
- Storage rack geometry

The total biases and uncertainties of the maximum k_{eff} needs to include the biases and uncertainties from both the primary code to secondary code validation, and the secondary code to benchmark experiment's validation.